OVERSIGHT

THE NUCLEAR REGULATORY COMMISSION RESEARCH PROGRAM

HEARINGS

BEFORE THE

SUBCOMMITTEE ON
ENERGY RESEARCH AND PRODUCTION

OF THE

COMMITTEE ON
SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES

NINETY-SEVENTH CONGRESS

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OVERSIGHT

THE NUCLEAR REGULATORY COMMISSION RESEARCH PROGRAM

TUESDAY, MAY 18, 1982

House of Representatives,
Committee on Science and Technology,
Subcommittee on Energy Research and Production,
Washington, D.C.

The subcommittee met, pursuant to notice, at 1:30 p.m., in room 2318, Rayburn House Office Building, Hon. Marilyn Lloyd Bouquard (chairman of the subcommittee) presiding.

Present: Representatives Bouquard and Lowery.

Staff present: Dr. Raymond Pennotti, professional staff; and Joyce Gross Freiwald, Republican technical consultant.

Mrs. **Bouquard**. The subcommittee will come to order.

Good afternoon, ladies and gentlemen. Today the subcommittee will begin hearings on the Nuclear Regulatory Commission's research program.

This $220 million program represents nearly half of the total NRC budget request. It encompasses research in a broad range of disciplines, from materials and structures to control systems and human engineering. This program makes use of large scale experimental facilities and complex computer simulations.

The management of such an extensive program is a difficult job. Numerous tradeoffs are required among diverse program elements in order to use resources efficiently. Few will argue with the proposition that these tradeoffs should be made using a systematic cost/benefit approach. One of the objectives of this hearing today is to gauge the management effectiveness of NRC's research program using this standard.

The interaction of the NRC research program with other efforts in both the public and the private sectors is complex, requiring consideration of issues relating to regulatory independence and duplication of effort. Nevertheless, the NRC, the Department of Energy, and all segments of the nuclear industry have unique contributions to make in the common goal of insuring nuclear powerplant safety. NRC research resources are, after all, limited as are all other budgets, and therefore these resources must be used efficiently. We will be examining in these hearings the appropriate interface of NRC's research activities with those of industry and the Department of Energy.

The subcommittee brings to these hearings a long history of energy research and development oversight, particularly in the area of nuclear power. We have repeatedly exercised this jurisdiction over the civilian nuclear programs of the Department of Energy, including those covering nuclear reactor safety, waste management, and advanced reactors, all of which have strong interfaces with NRC research. While this is the first time that we will have separate hearings on NRC's entire research program, we have been following with interest its growth and evolution for many years. We are particularly pleased to see the trend in the program toward more meaningful dialog with those outside of NRC pursuing related research.

We are very fortunate to have with us today the Director of NRC's Office of Nuclear Regulatory Research, Mr. Robert Minogue, to provide testimony on the NRC research objectives and program. We will also hear today from Dr. Chester Siess, who will represent the views of the Advisory Committee on Reactor Safeguards, which, among its other duties, evaluates the Commission's research program for the Congress. Dr. Siess has been a member of the ACRS since 1968 and served as chairman of the committee from 1972 to 1976.

I think the ranking Republican member of the full committee, Mr. Winn, will be here shortly. We will certainly welcome his opening remarks.

I would also at this time ask for unanimous consent, pursuant to House rules, that television and other recordings of today's hearings be permitted. I see there is no objection. So ordered.

Mr. Minogue, please proceed with your testimony. Without objection, you may proceed as you wish, and your entire testimony will be made a part of our record.

STATEMENT OF ROBERT B. MINOGUE, DIRECTOR, OFFICE OF NUCLEAR REGULATORY RESEARCH, U.S. NUCLEAR REGULATORY COMMISSION, ACCOMPANIED BY DENWOOD ROSS, DEPUTY DIRECTOR, OFFICE OF NUCLEAR REGULATORY RESEARCH, U.S. NUCLEAR REGULATORY COMMISSION

Mr. MINOGUE. My name is Robert Minogue. I am the Director of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. I am accompanied today by Dr. Denwood Ross, who is the Deputy Director. I am pleased to appear before your subcommittee to discuss the role of the Commission's research program, how it directly supports the Commission's regulatory mission, and how it impacts more broadly on improved safety of nuclear powerplants.

Our research program is part of the overall NRC program of regulation to insure the safety of nuclear powerplants and the associated fuel cycle; test and research reactors; radioactive waste disposal; and academic, medical, and commercial applications of radionuclides. The NRC performs its regulatory function by defining conditions for the use of nuclear power and materials and insuring, by technical review and inspection, that these conditions are met. The NRC research program provides technical information to aid in discharging these regulatory responsibilities.

The Office of Nuclear Regulatory Research was established by the Energy Reorganization Act of 1974, which listed the following functions for the office: First, developing recommendations for research deemed necessary for performance by the Commission of its licensing and related regulatory functions; and second, engaging in or contracting for research which the Commission deems necessary for the performance of those functions.

The Energy Reorganization Act was ultimately amended to include a new subsection, which stated: "The Commission shall develop a long-term plan for projects for the development of new or improved safety systems for nuclear powerplants." Within these legislative requirements and with Commission guidance, the objective of the Office of Nuclear Regulatory Research is to conduct the necessary research to develop a technical basis for sound decisions on regulatory issues and improved nuclear safety. This is being accomplished through the development of risk assessment methods for evaluating regulatory issues and their application to broad problem areas; second, improvement of the understanding of phenomena necessary for analyzing safety implications; and third, identification and definition of means to improve the consistency and coherence of the level of protection afforded by the NRC regulations.

To better coordinate research program planning, the Office of Nuclear Regulatory Research develops an annual long-range research plan that is submitted to the Commission for approval. The long-range research plan is a 5-year planning document that outlines programmatic approaches for research that will aid in the resolution of identified issues. Its primary purpose is to identify the principal areas or questions that require research in their resolution and to provide insights as to the level of future resource requirements. The process of developing a long-range plan is an effective vehicle for dialog between the Office of Nuclear Regulatory Research and the NRC program offices regarding the research directions needed to support the Commission's regulatory program.

We have just completed the development of the current update of the long-range research plan for fiscal years 1984 through 1988. I will summarize the major directions of the Commission's research program as laid out in that updated plan.

First, a significant effort is underway to improve the understanding of complex system transients and of the phenomenology of accidents involving fuel damage. This information should improve the understanding of needed plant design and staff operational capabilities in the event of such transients or accidents. This work is also expected to provide a more realistic quantification of fission product behavior, often referred to as the "source term," and of post-accident radiation levels to which in-containment equipment will be exposed. A better definition of the loads on containment from accidents involving fuel damage will also result from this work. It will also provide information for design of improved safety systems and mitigation features. The information that will be developed is expected to improve NRC's ability to implement its safety objective and to provide better information for best-estimate analyses and for use in risk assessments.

Second, a significant effort on hardware-related reactor facility safety research is now underway and will continue throughout the plan period. This work is investigating a wide range of problems, including neutron embrittlement of pressure vessel steel and pressurized thermal shock, the validity of nondestructive examination techniques, and the integrity of steam generator tubes and other parts of the primary coolant pressure boundary. The investigations also include evaluating the survivability of mechanical and electrical equipment in accident situations and studies of seismic safety margins and risks.

Third, emphasis will continue during the plan period on further improvements in methodology and on applications of probabilistic risk assessment techniques. We intend to apply these techniques to research planning efforts and to assess relative priorities of regulatory activities. Increased use of these methods will also be made by the NRC in the licensing review process. This effort will be augmented by increased use of system reliability analysis and evaluation, including the analysis of human error rate data.

Fourth, the planned phasedown of our loss-of-coolant accident/emergency core cooling system research, particularly the portion associated with large and intermediate breaks which was associated with the safety margins provided by the criteria of appendix K of 10 CFR, part 50, will continue and should be essentially completed in fiscal year 1985. This phasedown is reflected by reduced loss-of-coolant accident tests in our electrically heated thermal-hydraulic test facility, called Semiscale, and the planned termination of the NRC LOFT program. Note that the Department of Energy is spearheading an effort to organize an international consortium to continue the operation of LOFT. There is a shift of emphasis from large system code development to the application of codes for a better understanding of the progression and consequences of complex transients and accidents in both reactor and containment systems.

Fifth, research to improve the safety of operating facilities includes a significant program in the area of human engineering, particularly to improve the operator-machine interface. Research also includes evaluation of malfunctions of plant instrumentation, control, and electrical systems to determine the impact on safety. Survey meters and dosimetry techniques will be evaluated in order to improve occupational radiation protection.

Sixth, research on high-level waste management supports the NRC evaluation of the long-term performance of natural and engineered barriers to radionuclide migration. The natural barrier is the geologic setting, and the engineered barriers are the waste package, backfills, and shaft and tunnel seals. Research on low-level waste management also supports predictions of the long-term performance of the disposal system and includes research on waste package integrity, on characterizing decontamination wastes, on the impact of nonradiological toxic substances, and on geologic alternatives to shallow land burial. Research on uranium recovery waste management supports predictions of the long-term performance and safety of disposal techniques at active uranium recovery operations and remedial action at inactive sites.

Seventh and last, research in seismology and geology supports the operating reactor program, as well as the siting programs for reactors and high-level waste repositories. The program addresses particular problems of establishing the seismic hazard in certain areas of the Eastern United States.

In my discussion of the research program and its relationship to the regulatory process, I have explained the process used in developing our program and in setting priorities for its components. Integral to this process is an understanding of the relative risks associated with the operation of nuclear facilities along with their uncertainties, that is, the uncertainties in those risks.

The major part of the NRC research program is aimed at improving the ability to evaluate the performance of safety-related systems and to reduce the uncertainties associated with risk assessment. This improved ability is applied partly through research and partly through licensing staff review to identify sources of risk and to evaluate alternative approaches to risk reduction. Sometimes an improved understanding in an area of uncertainty leads to a conclusion that a potential source of risk, or the reduction of the uncertainty associated with a source of risk, is not significant and further research or planned regulatory action is not necessary.

In its early phases, the NRC research program was directed toward work needed to confirm regulatory decisions. The most dramatic example of this is the work on the efficacy of the emergency core cooling system, culminating in the LOFT program. Although this work can be called confirmatory, much of it relates to risk and to actual operating experience. For example, small-break loss-of-coolant accidents have turned out to be more common than expected and to entail system transients more complex than previously envisioned.

Our 1978 plan for research to improve the safety of light-water nuclear powerplants outlines a block of work that was based on relative risk. This plan began the work of examining the probability and effects of core-damaging accidents in light-water reactors and the means for either preventing them or coping with them. The Three Mile Island accident reinforced the need for this program, and it now receives one of the major shares of research funding.

Mrs. BOUQUARD. Mr. Minogue, I missed that. Pardon me. What program are you discussing?

Mr. MINOGUE. Madam Chairman, this is a program which we developed in response to the amendment to the Energy Reorganization Act. It required us to develop a long-range plan to lead to safety improvements, a specific program element that is not carried in our budget.

Mrs. BOUQUARD. You are referring to the safety program?

Mr. MINOGUE. This is the improved safety program, yes, ma'am.

The budget decision units for accident evaluation research, systems and reliability research, and facility operations can be traced directly to this program. Our research effort now has a major element called the severe accident research program, in which risk relevance is an integral part of evaluating and directing our effort.

I will go on to discuss the relationship with industry and the Department of Energy. We believe that the nuclear industry and the Department of Energy have a major role and interest in performing safety research and reliability research which is related to safety questions to insure that nuclear powerplants and other nuclear facilities are designed and operated safely and reliably. In support of this belief, there is an increasing level of cooperation and coordination among NRC, DOE, the Electric Power Research

Institute, the nuclear industry, and other countries in research program planning to insure that the appropriate level of effort is directed at resolving safety issues with the efficient use of resources and to prevent unnecessary duplication of effort.

In 1980, Congress passed the Nuclear Safety Research, Development, and Demonstration Act of 1980, to provide for an accelerated and coordinated program of light-water-reactor safety research, development, and demonstration to be carried out by the Department of Energy. The NRC staff has cooperated fully with the Department of Energy in its activities to implement this legislation.

The NRC viewed the legislation to which I referred earlier, the amendment to the Energy Reorganization Act, which amended our research charter, as an important extension to its mission. The legislation defined a role for NRC in coordination with the Department of Energy in improving the safety of reactors and nuclear fuel cycle facilities to supplement its ongoing confirmatory research mission, which was generally aimed at insuring the current level of safety.

The role of NRC under this legislation has been to obtain basic data and to develop analytical models to assess the feasibility and effectiveness of safety improvement and to specify regulatory requirements where implementation is desirable. The NRC then uses its expertise in risk assessment to develop and evaluate conceptual designs for improved safety systems.

Following the identification by the NRC of a concept for improved safety or reliability, a significant effort still remains in development work to determine feasibility. In this effort, we feel there is an appropriate role for the Department of Energy which could complement the NRC program. Examples of these functions include:

One, assess whether NRC's research solutions to a specific nuclear safety problem are commercially feasible; two, maintain and operate large research facilities, such as LOFT, that could have broader applications to energy research than the narrow nuclear-safety-oriented research objectives of NRC; three, provide a forum for dialog among parties concerning research programs designed to address current operating reactor plant problems with broad implications; and four, assist industry in finding or developing solutions to safety issues, particularly for future plants.

In addition to the NRC-DOE cooperation mechanism which I have described, NRC has extensive cooperative agreements and projects with industry groups such as the Electric Power Research Institute. Examples of some program areas where nuclear safety research is being coordinated with EPRI are: One, The release and combustion of hydrogen in containment buildings; two, fracture mechanics for reactor vessel and piping integrity; three, source term research dealing with the retention of fission products in reactor primary systems and representative model development; and four, integral test programs on a wide range of thermal-hydraulic transients in reactor systems.

In addition to coordinating with DOE and domestic industry groups, we also seek international cooperation whenever applicable. The quality of the research being done outside the United States, the availability of the results, and the cost savings through the use of unique experimental facilities here and abroad are all considerations on which we make decisions on our involvement in international programs.

Among our more important current agreements are ones with the Commission of the European Communities, the Federal Republic of Germany, France, Japan, the Nuclear Energy Agency, Sweden, and the United Kingdom.

The last topic, Madam Chairman, which your invitation to testify requested that I address was the impact of NRC safety research on actual safety in commercial powerplants. Research findings in the past have had as their primary concentration the confirmation of the conservatism of NRC safety standards. The loss-of-coolant accident/emergency core-cooling system findings are examples of this confirmatory effort. The research program has led to an increased understanding of the phenomena and systems behavior important to safety, and this knowledge allows the NRC staff to better evaluate the safety and conservatisms of designs and practices. It also provides the background needed to handle unanticipated problems when they arise. The codes which have been developed from this effort, and they are many, are widely used by industry in the analysis of operating problems, in operator training, and in efforts to improve plant availability.

Indirectly, this confirmatory effort contributed to actual safety by establishing benchmarks against which to measure future safety criteria. More directly, a portion of the research effort has led or is expected to lead to criteria for new regulations. Examples of such contributions are:

One, NRC-sponsored research has indicated that it may be appropriate to simplify our regulations on emergency core cooling. An additional benefit of the program is that the best-estimate codes developed as a result of this work will contribute to the capability of reactor training simulators to model accident transients.

Two, An active research program on developing and applying the techniques of probabilistic risk assessment to reactor safety questions is providing an essential capability to perform reliability and risk assessments and to establish quantitative safety goals.

Three, The LOFT experimental program has provided valuable insights on a number of emergency core cooling issues.

Four, The heavy section steel technology program has provided a significant portion of the basis for the nuclear pressure vessel codes of the American Society of Mechanical Engineers. This program has also developed evidence to support the presumption of a leak-before-break failure mode in reactor pressure vessels, and has generated substantial data that have been used in predicting irradiation embrittlement and margins to failure of reactor pressure vessels when flaws are present. It has also provided input to NRC requirements for fracture toughness of reactor pressure vessel materials, and has contributed data used in the ASME code which was referenced in appendix G to part 50 of our regulations, and has provided the data base for a regulatory guide concerning the effect of impurities on the sensitivity of steel to irradiation.

Quantifying the actual contribution to safety of these research results is very dependent on the refinement of risk assessment techniques and our ability to quantify plant safety. On this subject, the program is continuing to emphasize research on improvements in methodology and reduction in the uncertainty of probabilistic risk assessment techniques to support implementation of the Commission's safety goal.

This concludes my remarks, Madam Chairman. I would be glad to respond to any questions the committee may have.

[The prepared statement of Mr. Minogue follows:]

TESTIMONY OF ROBERT B. MINOGUE

DIRECTOR, OFFICE OF NUCLEAR REGULATORY RESEARCH

U. S. NUCLEAR REGULATORY COMMISSION

BEFORE THE

SUBCOMMITTEE ON ENERGY, RESEARCH, AND PRODUCTION

OF THE

HOUSE COMMITTEE ON SCIENCE AND TECHNOLOGY

MAY 18, 1982

TESTIMONY OF ROBERT B. MINOGUE
TO THE
SUBCOMMITTEE ON ENERGY, RESEARCH, AND PRODUCTION
OF THE
HOUSE COMMITTEE ON SCIENCE AND TECHNOLOGY
MAY 18, 1982

Madam Chairman:

My name is Robert Minogue. I am the Director of the Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES). I am pleased to appear before your subcommittee to discuss the role of the Commission's research program, how it directly supports the Commission's regulatory mission and impacts more broadly on improved safety of nuclear power plants.

Our research program is part of the overall NRC program of regulation to ensure the safety of nuclear power plants and the associated fuel cycle; test and research reactors; radioactive waste disposal; and academic, medical and commercial applications of radionuclides. The principal responsibility of the Nuclear Regulatory Commission is to ensure that public health, public safety, and the environment are adequately protected. The NRC performs this function by defining conditions for the use of nuclear power and materials and ensuring by technical review and inspection that they are met. The NRC research program provides technical information, independently of the nuclear industry, to aid in discharging these regulatory responsibilities.

The Office of Nuclear Regulatory Research was established by the Energy Reorganization Act of 1974 (Public Law 93-438). This Act stated "the Director of Nuclear Regulatory Research shall perform such functions as the Commission shall delegate including:

(1) Developing recommendations for research deemed necessary for performance by the Commission of its licensing and related regulatory functions.

(2) Engaging in or contracting for research which the Commission deems necessary for the performance of its licensing and related regulatory functions.

The Energy Reorganization Act of 1974 was amended by the Nuclear Regulatory Commission's Fiscal Year 1978 Authorization Act (Public Law 95-209) to include a new subsection, which stated:

> "The Commission shall develop a long-term plan for projects for the development of new or improved safety systems for nuclear power plants."

In response to this latter requirement, the Commission submitted to Congress a report dated April 12, 1978, entitled "PLAN FOR RESEARCH TO IMPROVE THE SAFETY OF LIGHT WATER NUCLEAR POWER PLANTS" (NUREG-0438). This report is updated each year via the NRC Annual Report.

Within these legislative requirements and with Commission guidance, the objective of the Office of Nuclear Regulatory Research is to conduct the necessary research to develop a technical basis for sound decisions on regulatory issues and improved nuclear safety. This is being accomplished through (a) development of risk assessment methods for evaluating regulatory issues, and their application to broad problem areas, (b) improvement of the understanding of phenomena necessary for analyzing safety implications, and (c) identification and definition of means to improve consistency and coherency of the level of protection afforded by NRC regulations.

Having briefly outlined the role of the research program, how it was established, and how it supports the overall Commission objectives, I will discuss the specific subjects you requested in your invitation to testify, which are:

- Mechanism for establishing priorities for program components,
- Relationship of NRC research to the regulatory process,
- Correspondence of funding levels to relative risk and known incidents in actual operating experience,
- Relationship of NRC research to nuclear safety research at DOE and in industry,
- Impact of NRC safety research on actual safety in commercial power plants.

Because the role of the research program is, as I have stated, to support the regulatory process, the setting of priorities for program components is dependent on their relationship to the regulatory process. To better coordinate research program planning, to assist the Commission in establishing appropriate priorities, and to ensure effective utilization of NRC resources in support of the Commission's regulatory mission, the Office of Nuclear Regulatory Research develops an annual Long Range Research Plan that is submitted to the Commission for approval.

The Long Range Research Plan is a 5-year planning document that outlines programmatic approaches for the research that will aid in the resolution of identified issues.

Its primary purpose is to identify the principal areas or questions that require research in their resolution and to provide insights as to the level of future resource requirements. The process of developing the long range plan is an effective vehicle for dialogue between the Office of Nuclear Regulatory Research and the NRC program offices regarding the research directions needed to support the Commission's regulatory program.

The review and comments the plan has generated have resulted in a shift in direction for the research program. Greater emphasis is now being given to research that addresses a broad range of technical information to support Commission decisions, including rulemaking activities. Previous emphasis had been on support or confirmation of continuing licensing activities. While these latter efforts will continue, the trend is toward increased use of the research program to provide information and methodology to support Commission decisionmaking and formulation of basic policy.

We have just completed the development of the current update of the Long Range Research Plan for fiscal years 1984 through 1988. I will now summarize the major directions of the Commission's research program.

1. A significant effort is under way to improve the understanding of complex system transients and of the phenomenology of accidents involving fuel damage. This information should improve the understanding of needed plant design and staff operational capabilities in the event of such transients or accidents. This work is also expected to provide a more realistic quantification of fission product behavior ("source term") and of post-accident radiation levels to which in-containment equipment will be exposed. A better definition of the loads on containment from accidents involving fuel damage will also result from this work. This effort will also provide information for design of improved safety systems and mitigation features. The information that will be developed is expected to improve NRC's ability to implement its safety objective and to provide better information for best-estimate analyses and for use in risk assessments.

2. A significant effort on hardware-oriented reactor facility safety research is now under way and will continue throughout the Plan period. This work is investigating a wide range of problems, including neutron embrittlement of pressure vessel steel and pressurized thermal shock, the validity of nondestructive examination techniques, and the integrity of steam generator tubes and other parts of the primary coolant pressure boundary. The investigations also include evaluating the survivability of mechanical and electrical equipment in accident situations and studies of seismic safety margins and risks.

3. Emphasis will continue during the Plan period on further improvements in methodology and on applications of probabilistic risk assessment techniques. We intend to apply these techniques to research planning

efforts and to assess relative priorities of regulatory activities. Increased use of these methods will also be made by the NRC in the licensing review process. This effort will be augmented by increased use of system reliability analysis and evaluation, including the analysis of human error rate data.

4. The planned phasedown of our loss-of-coolant accident/emergency core cooling system research, particularly the portion associated with large and intermediate breaks which was associated with the safety margins provided by the criteria of Appendix K of 10 CFR Part 50, will continue, and should be essentially completed in FY 1985. This phasedown is reflected by reduced LOCA tests in our electrically heated thermal-hydraulic test facility (Semiscale), and the planned termination of the NRC LOFT program. (Note that DOE is spearheading an effort to organize an international consortium to continue the operation of LOFT.) There is a shift of emphasis from large system code development to the application of codes for a better understanding of the progression and consequences of complex transients and accidents in both reactor and containment systems.

5. Research to improve the safety of operating facilities includes a significant program in the area of human engineering, particularly to improve the operator-machine interface. Research also includes evaluation of malfunctions of plant instrumentation, control, and electrical systems to determine the impact on safety. Survey meters and dosimetry techniques will be evaluated in order to improve occupational radiation protection.

6. Research on high-level waste management supports the NRC evaluation of the long-term performance of natural and engineered barriers to radionuclide migration. The natural barrier is the geologic setting and the engineered barriers are the waste package, backfills, and shaft and tunnel seals. Research on low-level waste management also supports predictions of the long-term performance of the disposal system and includes research on waste package integrity, on characterizing decontamination wastes, on the impact of nonradiological toxic substances, and on geologic alternatives to shallow land burial. Research on uranium recovery waste management supports predictions of the long-term performance and safety of disposal techniques at active uranium recovery operations and remedial action at inactive sites.

7. Research in seismology/geology supports the operating reactor program, as well as the siting programs for reactors and high-level waste repositories. The program addresses particular problems of establishing the seismic hazard in certain areas of the Eastern United States.

In my discussion of the research program and its relationship to the regulatory process, I have explained the process used in developing our program and in setting priorities for its components. Integral to this process is an understanding of the relative risks associated with the operation of nuclear facilities along with their uncertainties.

On the subject of risk, you requested that I discuss the "Correspondence of funding levels to relative risk and known incidents in actual operating experience."

The major part of the NRC research program is aimed at improving the ability to evaluate the performance of safety-related systems and to reduce the uncertainties associated with risk assessment. This improved ability is applied, partly through research and partly through licensing staff review, to identify sources of risk and to evaluate alternative approaches to risk reduction. Sometimes an improved understanding in an area of uncertainty leads to a conclusion that a potential source of risk, or the reduction of the uncertainty associated with a source of risk, is not significant and further research or planned regulatory action is not necessary.

In its early phases, the NRC research program was directed toward work needed to confirm regulatory decisions. The most dramatic example of this is the work on the efficacy of the emergency core cooling system, culminating in the LOFT program. Although this work can be called confirmatory, much of it relates to risk and to actual operating experience. For example, small-break loss-of-coolant accidents (LOCAs) have turned out to be more common than expected and to entail system transients more complex than previously envisioned. Thus, an undefinable but significant part of the confirmatory research funding can be ascribed to risk-related and experience-related researc

The 1978 "Plan for Research to Improve the Safety of Light Water Nuclear Power Plants," to which I have previously referred, outlines a block of work that was based on relative risk. This plan began the work of examining the probability and effects of core-damaging accidents in light-water reactors and the means for either preventing them or coping with them. The Three Mile Island accident reinforced the need for this program, and it now receives one of the major shares of research funding. The budget decision units for accident evaluation research, systems and reliability research, and facility operations can be traced directly to this program. Our research effort now has a major element called the Severe Accident Research Program, in which risk relevance is an integral part of evaluating and directing our effort.

Leaving the subject of the internal development of the NRC research program, I will address the subcommittee request regarding the relationship of NRC research to nuclear safety research conducted by industry and the Department of Energy.

We believe the nuclear industry and the Department of Energy have a major role and interest in performing safety research, and reliability research related to safety questions, to ensure that nuclear power plants and other nuclear faciliti are designed and operated safely and reliably. In support of this belief, there is an increasing level of cooperation and coordination among NRC, DOE, the Electr Power Research Institute (EPRI), the nuclear industry, and other countries in research program planning to ensure that the appropriate level of effort is directed at resolving safety issues with efficient use of resources and to prevent unnecessary duplication of effort.

In 1980, Congress passed Public Law 96-567, "Nuclear Safety Research, Development, and Demonstration Act of 1980," to provide for an accelerated and coordinated program of light-water-reactor safety research, development, and demonstration to be carried out by the Department of Energy. The Commission staff has cooperated fully with the Department of Energy in its activities to implement this legislation.

The NRC staff reviewed both a first and a final draft of DOE's response to Public Law 96-567 and, in both cases, provided comments to DOE. The approach being taken by the Department of Energy to draw up a comprehensive program management plan is reasonable and, if effectively implemented, should result in a program that will yield important safety information and designs that could significantly improve the safety of nuclear power plants. The NRC has provided staff personnel to serve on the DOE working groups that have been set up to assist the Department in formulating the plan and to review its progress and implementation.

As I previously mentioned, the 1978 NRC authorization act (PL 95-209) amended the Energy Reorganization Act of 1974 (PL 93-438) to require NRC to "develop a long-term plan for projects for the development of new or improved safety systems for nuclear power plants." The Conference Report stated that "the basic purpose of this research is the improvement of reactor safety and not the enhancement of the economic attractiveness of nuclear power versus alternative energy sources." Further, it mandated that the Commission "coordinate with the Department of Energy or other agencies which are conducting similar efforts and any actions taken to implement such a plan should take into account related activities in progress at other agencies."

The NRC viewed this legislation as an important extension to its previously established charter of confirmatory research. The legislation signaled an extension of the role of NRC in improving the safety of reactors and nuclear fuel cycle facilities to supplement its ongoing confirmatory research mission, which was generally aimed at ensuring the current level of safety.

The role of NRC under this legislation has been to obtain basic data and develop analytical models needed to assess the feasibility and effectiveness of safety improvements and to specify regulatory requirements where implementation is desirable. The NRC then uses its expertise in risk assessment to develop and evaluate conceptual designs for improved safety systems.

Following the identification by the NRC of a concept for improved safety or reliability, significant effort still remains in development work to determine feasibility. In this effort, we feel there is a appropriate role for the Department of Energy which could complement the NRC program. Examples of these functions include:

1. Assess whether NRC's research solutions to a specific nuclear safety problem are commercially feasisble;
2. Maintain and operate large research facilities such as LOFT, that could have broader applications to energy research than the narrow nuclear-safety-oriented research objectives of NRC;
3. Provide a forum for dialogue among parties concerning research programs designed to address current operating reactor plant problems with broad implications; and
4. Assist industry in finding or developing solutions to safety issues, particularly for future plants.

In addition to the NRC-DOE cooperation mechanism described above, NRC has extensive cooperative agreements and projects with industry groups such as the Electric Power Research Institute. Examples of some program areas where nuclear safety research is being coordinated with EPRI are:

1. The release and combustion of hydrogen in containment buildings.
2. Fracture mechanics for reactor vessel and piping integrity.
3. Source term research dealing with the retention of fission products in reactor primary systems and representative model development.
4. Integral test programs on a wide range of thermal-hydraulic transients in reactor systems.

In addition to coordinating with DOE and domestic industry groups, we also seek international cooperation whenever applicable. The quality of the research being done outside the United States, the availability of the results, and the cost savings through the use of unique experimental facilities here and abroad are all considerations on which we make decisions on our involvement in international programs.

Our international programs fall into four categories: (1) broad-scope bilateral general information exchange agreements, (2) bilateral research participation agreements, (3) multilateral agreements, and (4) narrow-scope specific studies. Among our more important current agreements are ones with the Commission of the European Communities, the Federal Republic of Germany, France, Japan, the Nuclear Energy Agency, Sweden, and the United Kingdom.

All of these cooperative programs with the Deparment of Energy, industry groups, or international interests are intended to maximize the use of NRC research funds and provide us with a more effective overall program.

The last topic your invitation to testify requested that I address is "the impact of NRC safety research on actual safety in commercial power plants."

Research findings in the past have had as their primary concentration the confirmation of the conservatism of NRC safety standards. The loss-of-coolant accident/emergency core cooling system (LOCA/ECCS) findings are examples of this confirmatory effort. The research program has led to an increased understanding of the phenomena and systems behavior important to safety. This knowledge allows the NRC staff to better evaluate the safety and conservatisms of designs and practices. It also provides the background needed to handle unanticipated problems when they arise. The codes developed from this effort are widely used by industry in the analysis of operating problems, in operator training, and in efforts to improve plant availability.

Indirectly, this confirmatory effort contributed to actual safety by establishing benchmarks against which to measure future safety criteria. More directly, a portion of the research effort has led or is expected to lead to criteria for new regulations. Examples of such contributions are:

1. NRC-sponsored reserach has indicated that it may be appropriate to simplify and improve Appendix K of 10 CFR Part 50. An additional benefit of the program is that the best-estimate codes developed as a result of this work will contribute to the capability of reactor training simulators to model accident transients.

2. An active research program on developing and applying the techniques of probabilistic risk assessment to reactor safety questions is providing an essential capability to perform reliability and risk assessments and to establish quantitative safety goals.
3. LOFT has provided valuable insights in quantifying the anticipated emergency core cooling delay caused by steam generation resulting from contact of the injected water with the hot vessel and piping metal; has demonstrated that natural circulation cooling and refluxing are viable methods of heat removal; has demonstrated that recirculation pump current is an indication of system conditions; and has simulated an accident in which a steam bubble formed in the pressure vessel. LOFT has also been used to study the desirability of requiring that the main coolant pumps be turned off or left on during a small-break LOCA. These results have had a significant impact on licensing decisions made by the NRC staff on a case-by-case basis.
4. The Heavy Section Steel Technology (HSST) Program has provided a significant portion of the basis for section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This program has also developed evidence to support the presumption of a leak-before-break failure mode in reactor pressure vessels, and has generated substantial data that have been used in predicting irradiation embrittlement and margins to failure of reactor pressure vessels when flaws are present. It has also provided input to NRC requirements for fracture toughness of pressure vessel materials as described in Appendix G to 10 CFR Part 50;

has contributed data used in the ASME code that was made part of Appendix G; and has provided the data base for Regulatory Guide 1.99 concerning the effect of impurities on the sensitivity of steel to irradiation.

5. Recent results from research on steam explosions has demonstrated a much smaller energy release than was previously assumed in the Nuclear Safety Study (WASH-1400). This is expected to have a significant impact on the probabilistic risk assessment of core melts and consequently on licensing requirements related to accidents resulting in severe core damange.

Quantifying the actual contribution to safety of these research results is very dependent on the refinement of risk assessment techniques and our ability to quantify plant safety. On this subject the research program, as I discussed earlier, is continuing to emphasize research on improvements in methodology and reduction in the uncertainty of probabilistic risk assessment techniques to support implementation of the Commission's Safety Goal.

This concludes my remarks. I will be glad to respond to any questions which the Committee may have.

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Robert B. Minogue, a native of Covington, Kentucky, did undergraduate work at the U. S. Military Academy, and at Thomas More College from which he received a B. S. degree in physics and mathematics in 1949. He also received an M. S. degree in nuclear physics from the University of Cincinnati in 1951. He is a 1952 graduate of the Oak Ridge School of Reactor Technology.

He began his professional career with Admiral Rickover, working from 1952 through 1956 on the Nautilus and Seawolf programs and on other early nuclear-powered submarine and aircraft carrier projects. He also participated in the design of the Shippingport reactor, the first commercial nuclear power plant. While in the Rickover group, he rose to the position of chief of the Nuclear technology Branch, with responsibilities for the sodium and water coolant chemistry program, shielding and radiological protection, and testing programs for land-based prototypes and shipboard nuclear power plants.

In 1957 Mr. Minogue joined the General Atomic Division of General Dynamics Corporation in the newly formed John J. Hopkins Laboratory for Pure and Applied Science, where he was head of the Design and Project Engineering Section for TRIGA, a research reactor. In this capacity, he was responsible for the engineering design and fabrication of a number of TRIGA reactors in the United States and other countries. In 1960, he transferred to the High Temperature Gas Cooled Reactor program as head of the Nuclear Plant Engineering Section. In this position, he was responsible for coolant technology, design of auxiliary helium systems and coolant purification systems, overall plant arrangement, shielding and operational radiological safety, and analyses of fuel cycle alternatives for the HTGR program.

He joined the Regulatory staff of the U. S. Atomic Energy Commission (now the Nuclear Regulatory Commission) in 1967 as Chief of the Special Projects Branch in the Division of Reactor Standards. Subsequently, he served successively as Assistant Director for Safety and Materials Protection Standards and as Deputy Director of Regulatory Standards, before his appointment as Director of the Office of Standards Development when the NRC was established.

In his position as Director of the NRC Office of Standards Development, Mr. Minogue directed the development of regulations, guides, and standards over the full range of NRC's regulatory activity; that is, in the areas of protection of the public health and safety, environmental protection, and safeguarding of nuclear materials. He is the U. S. member of the International Atomic Energy Agency's Senior Advisory Group on international nuclear reactor safety standards; and has served as a member of the Board of Directors of the American Society for Testing and Materials (ASTM). He is a Registered Professional Nuclear Engineer in the State of California.

In November 1980, Mr. Minogue was appointed Director of the Office of Nuclear Regulatory Research.

Mrs. BOUQUARD. Thank you very much, Mr. Minogue.

I would like to talk about the long-range research plan first of all. You refer to it as the first step in setting priorities. I want to better understand how the long-range plan helps the Commission establish the appropriate priorities, in view of the fact that last year's plan, as well as the new plan, has hundreds of pages of program-specific details but it really does not have any introductory discussion of the framework in which these individual program elements fit. It does not tell which part of the patchwork quilt they are going to fit into.

Mr. MINOGUE. Madam Chairman, that point is well taken. In fact, in the current edition of the long-range plan, we have taken a large step to correct the problem. The draft which we furnished to the committee staff is the prior version. I had much the same reaction to it.

Last year was the first year that we developed a long-range plan, and I think its greatest success was that it created a forum for dialog at the management level, between the Office of Research and the program offices of NRC. At that level, many of these long-term objectives were clear cut. A serious weakness of the plan last year, which was pointed out quite strongly by the ACRS, was that the lack of this kind of connective tissue and general discussion of the program objectives in broader terms made it hard for other people to use and react to the report.

In the final version of this year's plan, I think we have taken some steps to correct this. We will do better next year, and certainly the version we furnished to the staff was deficient in this respect.

Mrs. BOUQUARD. I do not want my remarks to be taken as having a critical nature but as constructive dialog to have an effective long-range plan. I notice also that there were no specific resource requirements for the program. I cannot understand how NRC can insure effective utilization of these programs if they do not have pricetags included with it.

Mr. MINOGUE. We do have a companion document, which I will be glad to furnish to the committee, which was used by the Commission and by the program office staff, that gives an approximate level of the resource requirements. We did not bind it into the final report because it does contain preliminary budget information, and this would lead to some restrictions in its distribution. But there is a companion document that provides that information.

Mrs. BOUQUARD. I notice that the Nuclear Safety Oversight Committee's reactor safety research review group suggested that the plan should really be restructured in an effort to avoid automatically doing more of what we have been doing all along; and also should be more specific in defining the objectives of the programs as well, what we can actually deliver as a result of the programs that have been specified. Would you agree with that?

Mr. MINOGUE. Yes, Madam Chairman, I do agree with that. That is one of the hard things of managing a program of this type, when the agency has a relatively narrow mission. I tried to really emphasize that in my testimony. Our mission is fundamentally concerned with supporting the regulatory program, and that is a relatively narrow base for research. If this mission requires owning and operating complex research facilities with broad capabilities, there is a problem for us with these facilities to generate a fully cost-effective program, because we do have a narrow base. Very often we get the answers we need and, ideally, could move right out. But once built, facilities are built and tend to generate programs, particularly if they are good facilities with broad capabilities. I touched on that in my testimony when I suggested a role for the Department of Energy in the operation of multipurpose facilities on which NRC could buy experimental work and so could other users. I think this would go a long way toward eliminating this very real tendency for programs to perpetuate, which I think is a well-taken point of the Oversight Committee's review group.

Mrs. BOUQUARD. On page 7 of your written testimony, I notice you said the understanding of relative risks and uncertainty plays an integral part in assessing priorities in the research program. But then on page 8 you said that a major part of the research program should be aimed at improving ability to reduce the uncertainties associated with risk assessment. I would appreciate it if you would clarify that for me.

Mr. MINOGUE. Yes, Madam Chairman. This is a case of which comes first, the chicken or the egg. We have been able to use these techniques fairly successfully in cross-comparing research priorities within some narrow area where we are basically dealing with the same problem. We have been much less successful in using these techniques, because the risk assessment techniques themselves are not sufficiently mature, when we reach across broad areas of safety and begin to compare things that do not have a clear correspondence of being in the same set.

I think the direction toward greater use of these techniques is clear. I know the ACRS feels that we could go further, faster. All I can say is that over the last several years, we have made a lot of strides in developing risk assessment techniques sufficiently to use as a priority setting tool, and I expect this progress to continue. It is an ongoing process, but right now we are somewhat in a circular mode. Your point is well taken.

Mrs. BOUQUARD. Returning to the nature of the research, the NRC Advisory Committee on Reactor Safeguards, as well as the NSOC Reactor Safety Research Review Group, commented on the limitation of NRC to confirmatory research. They felt that this should be relaxed. In your testimony, you stated that you felt the NRC should have a more aggressive role in improving reactor safety. Do you look upon these roles as compatible? Are they integrated? Do they cover both objectives?

Mr. MINOGUE. Madam Chairman, we certainly have both objectives, even in the current program. I have been in this position roughly 2 years, but under my predecessors it is a fact that there was undue emphasis on confirmatory research and that narrow scope, I think, made the Office of Research less effective than it could have been. We have been very solidly backed by the Commission in efforts to change this trend. Now I think the primary emphasis is definitely not on confirmatory research but on dealing with broad safety issues directly. But there are still elements in the program, and I think there should be, of some confirmatory research.

Mrs. BOUQUARD. You talked about the work you are doing to improve the safety of light-water nuclear plants since the Three Mile Island accident. How much funding does that work represent, Mr. Minogue?

Mr. MINOGUE. A lot depends on exactly how you break it down. It draws elements from three decision units. I have here before me a breakdown that we made up some months ago. With a relatively broad definition of the parts of the program that relate to improved safety—that is, the definition that includes elements of assessing phenomenology to provide a design basis for improved safety systems—we have a calculated rate of expenditure in fiscal 1983 of $22.6 million. I can get a bigger number if I define it more broadly, and I can get a smaller number if I define it more narrowly. But the midrange level is $22.6 million for 1983.

It is a program that is growing. In the prior year, 1982, it was $19.4 million and $11.9 million in 1981.

Mrs. BOUQUARD. That is the middle-of-the-road funding that you would attribute to the safety program?

Mr. MINOGUE. Yes, ma'am. That is a midlevel definition. You can define it very broadly or very narrowly.

Mrs. BOUQUARD. You said you were cooperating with DOE, working out a comprehensive reactor safety program. Are you going to contribute to this program in areas where you already have a strong program?

Mr. MINOGUE. This is a key question, because the DOE budget in this area is quite small. I think there are a lot of things that they could do that are not being done now. I listed some in my testimony. I think there are some things that we do that they could do better.

Mrs. BOUQUARD. Could you elaborate on some of those?

Mr. MINOGUE. Yes. Any time you begin to get into a mode—Let me back up, if I may. What we do best is to assess basic safety problems and the basic data, the phenomenology: how accidents occur and how the fuel behaves. When you begin to say, let us turn that into a commercially viable product, a new system design or a new type of instrumentation, that is not our strength. That type of thing requires a knowledge of industry practices and a level or interface with industry that we lack. Of course, a lot of that kind of work is done by private industry, and we interface well with them in that area. But I think a lot of that type of thing could be done by DOE. If they had an effective program in this area, there are parts of our program where I think we would want to reexamine where we have drawn the line, where we stop doing what we do. It is clear to me that our strength is in assessing these basic safety questions, and we are not very good when it comes to doing commercial designs.

Mrs. BOUQUARD. You feel that there is DOE research, research appropriate to you, and research appropriate to the industry. Do you have a memorandum of understanding with DOE to address these distinctions and the way in which we can most effectively use our research funding?

Mr. MINOGUE. Yes, ma'am. We have a memorandum of understanding with DOE, and there are periodic coordination meetings. The program basically has only limited funding.

Mrs. **Bouquard**. I was very happy to read in your testimony that you do have an increasing level of coopration with industry. I think most of your efforts with EPRI are really more hardware oriented. Are you depending on industry to supply you with information concerning operations and maintenance limitations? Would you be a little more specific on your industry interface?

Mr. **Minogue**. Yes, Madam Chairman, I will. There are several examples I would like to give. I think probably the most important is, in the last year-and-a-half, we have executed or are negotiating a number of agreements with industry, either with EPRI and/or with the manufacturers, where they will go together with us on the construction and operation of the research facilities. This speaks to the problem I discussed earlier of it not being desirable for us to be the sole owner. It enables us to significantly reduce the cost of the program to the taxpayer, because the construction cost is spread over a number of organizations.

Another area that I would like to single out—and I think this is particularly true of EPRI—we have been working toward what really will amount to a consolidation of program planning. We are not there yet, but we have made a lot of progress. The ultimate goal is that when problem areas are identified, we will work together. In many of these cases DOE is involved. I should make that clear. We will work together with the industry people and reach an agreement on what information or data is needed, how it all fits together, reach some understanding as to whether it is more appropriate for some part of it to be done by us or by the industry people. So we end up with an overall research program that is really coherent, that makes a set, where the results will fit together in the last analysis and be usable by all parties and where the costs will be much reduced.

The experience we have had in the areas where we have been most successful with EPRI would indicate to me that this may well reduce the cost of the program, not only for us but for the industry. It is just more careful attention to how the programs fit together.

Mrs. **Bouquard**. Can you think of any changes that could be made to make this interface with industry more effective?

Mr. **Minogue**. I think the main element here is a matter of a commitment on both sides to work together, to make this work. We have had that commitment. Last spring, last year, this is a matter that our Commission focused on very specifically and I have their full backing. All of the commissioners have backed us on this, that we can work with industry, in terms of a consolidation of program planning. I think it is necessary that there are some safeguards here that we do not compromise our regulatory independence. It has been easy to deal with. But what really was needed I think now exists, and that is the commitment on the part of the Commission to support the staff in these efforts. In past years, the director, the person who had my job, would have found it difficult to agree to some of the business arrangements that we have made successfully with industry that have not in any way involved any compromise of our regulatory independence.

Mrs. **Bouquard**. Do you see this commitment to cooperation in industry as well?

Mr. MINOGUE. Yes, ma'am. It is there. Like anything else, institutions take on a life of their own. It is very hard to turn something like this around. I would not pretend that we are all the way there, but we have made a lot of progress.

Mrs. BOUQUARD. We are working toward that goal?

Mr. MINOGUE. We are working toward a goal, and at the management level it is very clear to me that, say between us and EPRI, Mr. Taylor and I are after the same goals. At the staff levels sometimes it is a little hard to make these things work. Sometimes it is a little uncomfortable, because it does involve at all times this element of providing safeguards. We have to maintain independence. We have to do what we do in an open environment. We have to release the data. The public has to be able to see what we are doing. That creates some institutional problems that I think we are able to deal with, but it makes it hard to do it overnight. It has been a slow, steady progress, I think.

Mrs. BOUQUARD. Very good. I am glad we are working in that direction. I have been rather concerned on some of the regulations. I think there is a widely held perception among our reactor safety researchers that some of the results of our research do not lead to indicated changes in regulations, especially when they show that some of these regulations may be overly conservative. Are we really taking the impact of our research and putting it into practice to enhance the impact of research results on regulations?

Mr. MINOGUE. I think the biggest impact of research work on the regulatory process, as opposed to regulations, has been in the widespread use of the codes that we have developed—that is really our major product—by both the industry and the regulatory staff. It is a fact, though, to respond to your direct question, that the ability to see some of these research results fully incorporated into the regulations has been less than completely satisfactory.

Part of this, I think, was problems we have now dealt with of a difficult interface between the regulatory part of the NRC staff and the research part. Under the Atomic Energy Commission, these were quite separate entities. It was fairly difficult to mesh the two organizations. I think they now are solidly merged. That has been one of the problems.

I think the second element here has been a concern that the regulatory staff has had with what I will call piecemeal revision of the regulations. They are more willing when you have a complete body of data that goes over a whole range, and it is clear what the overall conservatism is over the whole problem, then you can talk about modifying the regulations. A good example of that would be the work that I referred to in my testimony that has supported the ASME boiler and pressure vessel code. That is in the regulations, because there is that broad coverage across the whole range. But this has not been true, though, for some of the problems relating to the emergency-core cooling criteria, where I think the perception has been that we are not ready yet for a broad revision of the regulation. We may be able to make and are able to make some simplifications in the regulations, but for a broad review, there is still more data required.

Mrs. BOUQUARD. You are referring to appendix K?

Mr. MINOGUE. Yes, ma'am.

Mrs. BOUQUARD. Is research continuing in this area as well? When will it be completed?

Mr. MINOGUE. The work dealing with leading to the large loss-of-coolant accidents is in its final stages. I gave the completion date in the testimony; I believe it is 1985.

Mrs. BOUQUARD. How much money are you spending on this effort, Dr. Ross?

Dr. ROSS. Most of our large-scale work, and particularly the part with the cooperative agreements abroad, together with the assessment of some of our more complicated and complete computer codes, should be done by then. I think the final stage of changing the regulations could certainly be done then.

Mr. MINOGUE. I would like to add a comment to this, Madam Chairman. I think it is important in this context. I find, in talking to people in industry, that the people out there who are going to meet the regulation are not eager to change it today, in terms of a broad and sweeping review of appendix K. I think the consensus there is, let us wait and get these additional results and do it right. They are concerned about starting off to do it now——

Mrs. BOUQUARD. It might be a piecemeal approach?

Mr. MINOGUE. In a piecemeal way. What I referred to in the testimony was more in the nature of simplifying some of the requirements in the present regulations for reanalysis every time there is a minor design change. I think we have the basis to do that today. But a broad, sweeping revision of appendix K is several years away.

Mrs. BOUQUARD. Back to what you were talking about, will we put that into action, the changes that you were referring to?

Mr. MINOGUE. Yes, ma'am. The staff is well along in drafting a revision to appendix K that would provide some simplification of the requirements. It is not a substantive major change, across the board; it is more in the nature of a simplification.

Mrs. BOUQUARD. That is very good. Let us turn to the loss-of-fluid test facility, the LOFT facility. ACRS and NSOC have commented that this facility is no longer cost effective in ensuring or improving reactor safety. You have responded to the criticism by phasing down support of the facility. Then you are prepared to help DOE support the facility for research with other nations. What kind of research do you feel we should do in this facility that would warrant this support?

Mr. MINOGUE. We have a number of tests that have been identified by the reactor licensing people, which are primarily in the nature of additional transient tests. These are tests which, if we were the sole owners and operators of the facility, would not be cost effective, and we would not propose to do them. If the funding base of the facility were spread so that we would pay only a small cost, the work would be of sufficient value, in the opinion of our licensing staff, to warrant our participation at a level that has been estimated at basically $5 million per year to provide funding directly related to these tests that we have identified. Your staff has been furnished a larger number for NRC support that includes some negotiations regarding decommissioning costs and things of that type that are not related to testing. So the only testing charge that we would foresee if the international consortium should succeed would be $5 million per year, which is a price level that is felt by the licensing staff to be cost effective for the tests that they propose.

Mrs. BOUQUARD. I believe that in a letter to Mr. Fuqua, you said that DOE and NRC would spend about $15 million.

Mr. MINOGUE. The figure of $15 million has been discussed. That breaks down as follows: It is $5 million from us for testing; it is $5 million per year for 3 years, or a total of $15 million, for a hold-harmless clause on decommissioning charges against NRC; and then $5 million per year from the Department of Energy for testing. So it is a total of $15 million per year, for the next 3 years.

Mrs. BOUQUARD. But what part of the $15 million is NRC's?

Mr. MINOGUE. Of the $15 million per year, we would put in $10 million.

Mrs. BOUQUARD. Yours is $10 million, and DOE's would be $5 million?

Mr. MINOGUE. Yes, ma'am.

Dr. ROSS. Madam Chairman, we should point out that this is only about half of what it takes to run the facility. If the facility is to be run at all, then presumably the international community would have to make up the other half. Also, one of our $5 million was contingent on the tests that were deemed necessary would in fact be run, and that decision has not yet been reached either. So that half of our $10 million was contingent upon getting the right tests.

Mrs. BOUQUARD. I understand that this facility was established to examine large break loss-of-coolant accidents; is that correct?

Mr. MINOGUE. Yes, ma'am.

Mrs. BOUQUARD. If you are to conduct scientifically meaningful experiments there, would it require a lot of modification of the facilities?

Mr. MINOGUE. The facility has severe problems of scaling and of atypicality. I do not think it is possible to modify it to overcome those problems. It is more a matter of devising tests where you can run them in a way that by comparing data from other facilities and by very careful attention to the planning of the program, you can get results that you can interpret and believe. But I do not think it is possible to modify the facility.

Mrs. BOUQUARD. So there will not be modifications at all of the LOFT facility?

Mr. MINOGUE. I cannot really speak for the plans of DOE, in terms of the consortium. Certainly, last year when we were examining the question of whether to terminate operations or not—and we did decide to terminate in February of next year—one of the matters that we examined—Dr. Ross headed a special review group that the Commission established—was the problem of how do you deal with this atypicality and how do you deal with these scaling problems. It is clear, there are major problems with the facility, and it is very difficult to use the results without real care and without a lot of data from other facilities. We do have other smaller facilities that provide data as well.

Dr. ROSS. May I add, the money I was speaking of assumed no modifications. For roughly $30 million, the facility is run as is. The tests we are doing this year, and in particular early 1983, we will be doing tests that involve heating the zircalloy up to about 2,000 degrees Fahrenheit. You only do that once, and then you throw the core away. This is an expensive time-consuming process. There would be no such tests in the consortium for the $27 to $30 million we are talking about. If anything like that gets done, it will take a lot more money.

Mrs. BOUQUARD. It would take a lot of cooperation to make it anywhere near cost effective?

Mr. MINOGUE. Yes, ma'am. It would require a lot of funding from other sources. Whether that funding will materialize or not is highly speculative.

Mrs. BOUQUARD. I did not mean to spend so much time on that area.

Turning to seismic safety margins research program, what is the overall cost of that program?

Mr. MINOGUE. This program, I believe the cost over its life has been in the neighborhood of $15 to $20 million. I will have to double check that, and if that is not correct I will submit a revised figure.

The program as originally contemplated was more concerned with quantifying seismic risk and being able to deal with the fact that a lot of plants were designed many years ago for lower levels of earthquakes than are now seen as likely. Since there are large margins in these designs, there was an effort to decide how much margin there was in a normal design to deal with higher values of seismic input.

More recently, we have added an objective to this program that I think is long overdue, which is to look at the balance between seismic design requirements as we put them on and the safety of the plant in other respects, the rigidity that is introduced by many of the design requirements. That is a program that is not part of the seismic safety margins research program that was started just a year or so ago and may yield some very important results over the next few years.

Mrs. BOUQUARD. How important would you say that modeling uncertainties are in the program?

Mr. MINOGUE. That is a difficult question to answer because the facilities are very complex. If you could be absolutely certain that you had completely accurate, as-built drawings and that you had exactly characterized the facility, there are ways of dealing with the modeling uncertainties. There is a wide variation between what is built and what is on the drawing. I would say this has been a very difficult problem.

Mrs. BOUQUARD. In other words, it would take a lot of additional research in this area before conservatism in the regulations could be determined in this area?

Mr. MINOGUE. I would hope that we could come up with a better way of doing it. The basic design approach that was adopted some years ago was to make the systems more rigid with additional constraints. Of course, this makes them more complex and makes them difficult to model. I really feel there has to be a better way of doing this. It would be of benefit really only on new plants. We are hopeful we could come up with an approach to the response of these systems in earthquakes.

There is a tremendous ability in the plant to absorb earthquakes without damage. If we can take that into account better, I think we may be able to significantly simplify the design requirements, which will make the plant simpler.

Mrs. BOUQUARD. In the long-range research plan that you are referring to, you said that a great deal more work was being done on the advanced reactors by industry and DOE. You are not putting the same intensified efforts into the advanced reactors as you are in the LWR's; is that correct?

Mr. MINOGUE. Yes, ma'am. That is correct. There are two reasons for that. You have touched on one, which is a feeling on our part that in the regulatory process we should give more attention to what an applicant should do to support his own application. We have been much more conscious of what we should do and what the applicant, in this case DOE, should do. The second reason, and I think it is very important, is that a lot of the work necessary to evaluate the Clinch River project has already been done. In a prior era, there was a lot of research done by the NRC staff—that is by laboratories on our funding—that have dealt with a very large percentage of the research issues necessary to deal with the Clinch River project. It is a combination of those two: a feeling that a lot of work has already been done, and a feeling that of what remains to be done, much of that should really be done by the Department of Energy. That has led to this relatively small amount of money in the budget.

Mrs. BOUQUARD. So you feel that most of those issues have already been addressed in previous work done by NRC in reference to Clinch River?

Mr. MINOGUE. Yes, ma'am. I think the research issues have largely already been dealt with.

Mrs. BOUQUARD. So this is really part of a trend of coordinating your programs more with industry to avoid duplication in efforts?

Mr. MINOGUE. Let us say that is one way we do it. The same applies in the waste area. We have taken a much tougher attitude in the last few years, as budgets are tight and we have to be more careful and more businesslike.

Mrs. BOUQUARD. In other words, you are working to avoid any duplication in research on breeder reactors or high-level waste management that is done in other organizations?

Mr. MINOGUE. Yes. We are putting more emphasis on putting ourselves in a position where we can evaluate and use the research results done by the applicant, rather than try to do the work independently. That is not to say there would not be some occasional intentional duplication as a form of verification, but that is not the norm.

Mrs. BOUQUARD. But you would review the research work that was done by DOE or industry?

Mr. MINOGUE. Yes, ma'am. Or we would do research that really would be aimed at assessing the errors, the margins in his research work, at least in a narrower program.

Mrs. BOUQUARD. Another criticism made by the ACRS was that the accident evaluation and mitigation decision element was not laid out to answer specific questions that would require answers and, in fact that are answerable. Have you made any changes in that area as well?

Mr. **Minogue**. Yes, we have. That criticism, I think, was very well taken. That program, as first laid out, had too many elements of, "Here are the facilities that we have. Let's see how we can use them." I think we have made a number of strides, and we are still working on it, to develop what I have called a matrix of information needs: what are the questions this effort is trying to answer, what data are necessary; and try to design the experimental program to look at the available facilities and figure out what the most cost-effective combination is to get the needed data. We are still working on that. We have a meeting scheduled within the next few weeks with an ACRS subcommittee to present to them our progress to date. I think we have made a lot of progress, but we are not there yet. This program is very difficult, because we are trying to avoid building new facilities, which is costly. It is very difficult to figure out how best to use what you already have and the right cost-effective mix of facilities that are located in various laboratories around the country.

Mrs. **Bouquard**. You have been very kind and very responsive witnesses. I wanted to get into the program focus a little bit, but since we have gone well into an hour with you as a witness, I would like to submit some questions that I would like to have submitted for the record. If you would be so kind as to respond to our questions, we will certainly relieve you of this responsibility at this time. Thank you very much.

Mr. **Minogue**. Thank you, Madam Chairman.

Mrs. **Bouquard**. Thank you both for your excellent testimony.

Our next witness is Dr. Chester Siess, who is Chairman of the Nuclear Safety Research Committee of the Advisory Committee on Reactor Safeguards.

We are certainly happy to have you with us today, Dr. Siess. We look forward to your testimony. We do have your written testimony, and you may proceed as you wish. You may summarize, if you like, and your entire testimony will be made a part of the record of this hearing.

STATEMENT OF DR. CHESTER P. SIESS, CHAIRMAN, ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SUBCOMMITTEE ON THE SAFETY RESEARCH PROGRAM

Dr. **Siess**. Thank you very much, Madam Chairman.

I have prepared some remarks which, for the most part, will not repeat what was submitted in the written testimony. I would like to make some remarks of a general nature, which will relate to some of the areas of interest that were identified in your letter of April 30, and then I will be happy to try to answer any questions that you may have about the NRC's safety research program and the comments that the ACRS has made on it.

The ACRS has always been interested in research on reactor safety. Our interest precedes the formation of the NRC by several years. But our formal review and evaluation for the Congress goes back only 5 years. As you know, we have prepared five reports to the Congress, each of which has addressed the current or proposed

NRC safety research program in considerable detail. Although these reports were prepared by the ACRS as a collegial body, and thus represent the consensus of a 15-man committee, the basis for the reports has been a detailed review of the various elements of the research program by each of several specialized subcommittees of the ACRS. These subcommittees include the members most knowledgeable about the program areas being reviewed, supplemented and assisted by various expert consultants that serve the committee.

I mention this to make it clear that, although my job is to coordinate the review and report on the research program, I am not necessarily expert in all areas. There will undoubtedly be some questions, or many questions, that I cannot answer. If this is the case, we will be happy to try to provide you with answers in writing some time later.

I would like to comment briefly on three areas in which you indicated an interest: Priorities, who does what or who should do what, and how effective the program has been.

First, priorities and how they are established. I do not believe that anybody in the NRC, in the ACRS, or in the industry has ever been satisfied, either with the priorities that have been established or how they were arrived at. I do not expect to see the day when any of those groups are completely satisfied. It is much too complex a problem. The ACRS has always differed with the NRC staff on priorities in some portion, perhaps 25 to 30 percent, of the research program. We communicate these differences to the staff through our reports to the Congress and reports we make to the Nuclear Regulatory Commission itself, and in numerous meetings with the NRC staff. We advise them of our concerns about the program in general and sometimes in detail.

In a great many instances, the staff has been responsive to our concerns and has accepted our advice, but there are always some areas where they disagree or where we think they are not doing enough or not doing it fast enough. I believe Mr. Minogue acknowledged that difference. There are various reasons for these disagreements, but in most cases I believe they are due simply to differences in perspective or objective.

The NRC's Office of Nuclear Regulatory Research is still constrained to emphasize confirmatory research or research in response to user needs. These are not always the same, but they are usually pretty much the same. The NRC is committed, by guidance from the Commission, to give priority to operating reactors and to light-water reactors. On the other hand, the ACRS is concerned less with licensing and more with safety, if you care to make a distinction between licensing and safety. I know people have. We are concerned with research to improve safety of new reactors, of future reactors, as much as we are with the safety of currently operating reactors. We have had a long interest in research to improve safety, and we have always supported the programs to achieve that. There are such programs, as Mr. Minogue mentioned, but they still represent a small part of the NRC's research program.

I should pause here and say something I think is important. The NRC research program is changing. It is changing in approach. It is changing in philosophy. It is changing in content. I think it has been changing over most of the 5 years we have been looking at it in detail. It certainly has been changing in the last couple of years. So some of the things I am saying represent a little bit of the past. There are some trends in the future. I think research to improve safety, as I can identify it, is increasing. I think there are changing attitudes, and there is some change in the direction that is coming from the Commissioners.

I would like to talk briefly about who does what, in terms of research, industry, NRC, DOE, and so on. The ACRS has always supported strongly the need for a safety research program that is controlled and administered by the agency responsible to the public for the safety and regulation of reactors. In the early 1970's or even before that, when all of the safety related research in the old Atomic Energy Commission was carried out by the Division of Reactor Development and Technology, the ACRS argued before the Atomic Energy Commission that it was not appropriate for all safety research to be done by the same division involved in development and promotion efforts.

We felt that there was a difference between the research needed to make a facility safe and the research needed to know it was safe. We were not very successful in those arguments until 1973, when the safety related research was taken out of DRDT and assigned to a new Division of Reactor Safety Research. That division was still responsible to the AEC General Manager, which was the promotion side of AEC, rather than to the Director of Regulation, which was the regulatory side. But it was separated clearly from development and promotional efforts.

The first Director of Reactor Safety Research was Herbert Kouts, and I believe you are hearing from him tomorrow.

When the NRC was established in 1975, this program of safety-related research was transferred from AEC to the NRC, and of course then became totally under the regulatory arm. The Division of Reactor Safety Research essentially became what we now call the Office of Nuclear Regulatory Research in the NRC, still with Dr. Kouts as head of it.

We are generally quite happy that NRC has the responsibility, the funds, and the capability to do research related to safety. But this does not mean that NRC can or should do all of the safety-related research, or that it must rely only on its own research to answer questions, write regulations, or to establish policy, nor has this been the case in many instances. Again, there have been significant changes in the attitude of the Commission and the research people toward research by others in the last 2 years especially.

Much safety research has been done by industry, either on its own or as required by the NRC. They do a lot of research simply to answer questions that the NRC has asked.

Mrs. BOUQUARD. Doctor, do you feel that NRC should duplicate the safety research that is done by industry?

Dr. SIESS. I do not think they should duplicate it. They have to do whatever is necessary in order for them to be able to accept it. That is, you do not necessarily have to duplicate it. You may have to monitor it. You may have to examine it or follow it very closely.

But you have to be able to accept whatever somebody else does with the same level of confidence as if you had done it yourself if you are going to use it in regulations.

Mrs. **Bouquard**. The point I was trying to make is, you feel that the research done by industry should be examined by the NRC but their research efforts should not necessarily be duplicated?

Dr. **Siess**. For example, the various elements of the industry submit to the NRC results of research in the form of topical reports. Not all topical reports are research, but many of them are. The NRC may have questions about them, and eventually, after some give and take and maybe some additional work, they will accept a topical report. They will accept those results and then the information becomes as useful, or as used, as if NRC had done the research themselves.

Theoretically, the NRC can force the industry to do just about anything it wants. It can ask questions and tell the industry, "You are not going to get a license until you can answer these questions," and the industry goes out and does the research. This is not always done; sometimes it is. I think the pressure relief valve testing program is a fairly good example. I do not know whether you would call that research or testing; the line is not all that clear. NRC said, "We have to have some information on this"; industry went out and got it. It has been done in such a way that NRC eventually will be able to accept it.

There is a lot of safety research being done in other countries, much of it by regulatory agencies with the same degree of credibility as NRC. A lot of that is being done cooperatively with NRC and some in other ways.

DOE is doing a lot of safety-related research, but the vast majority of it is on the breeder reactor. DOE has done practically no safety research on light-water reactors since NRC was spun off the old AEC. When the Division of Reactor Safety Research went to NRC, that essentially killed all light-water reactor safety research in DOE. The program for research to improve safety was supposed to provide some funds to DOE, but I am not sure it ever got funded. They had a small amount of money, but it was $10 million at the most. We looked at it occasionally. I do not know what they have now but it is a negligible amount.

The Commission has taken a stand, as I believe Mr. Minogue said, to recommend the consolidation and coordination of programs with industry and other agencies, and the Commission is supporting the Office of Research in that type of work. We, of course, have made similar recommendations in a number of areas.

As I mentioned, during the past 2 or 3 years we have seen a signficantly increased effort by NRC to cooperate with and utilize the research of others. This has been driven, I think anybody will admit, simply by the fact that the research budget has not been increasing as fast as it has in the past. If you are going to answer questions, you are going to have to get help from other people. We believe that this direction is quite commendable, and we think it should continue.

Nevertheless, we continue to believe that a substantial amount of safety-related research and a major portion of the current program must continue to be done by the NRC in order to supply a sound basis for its safety requirements and to maintain both the reality and the appearance of independence. There is no substitute for doing a certain amount yourself so that you are absolutely sure it is good.

I would like to say a little bit about the cost-effectiveness of the NRC research program. Cost is no problem; we know what it costs. It is a lot of money. It has always looked like a lot of money to me because I worked for a university.

Mrs. **Bouquard**. You do not see that kind of funding coming into your research, do you, Doctor?

Dr. **Siess**. No, but it is about one-tenth of the cost of a 1000 megawatt nuclear plant, so it depends on which way you look at it.

Its effectiveness is a little difficult to evaluate, because, first, we have to know what the purpose of the program is to decide how effective it was in achieving that purpose. The purpose of the NRC research program has never been all that clear.

In the Commission's policy and program guidance for 1982, they said that the purpose of the research program is to assist in establishing regulations for existing and future facilities. By this measure, I do not think it has been very effective, because I cannot point to an awful lot of existing regulations that have resulted directly from the research program.

Congress, when it established the NRC and authorized it to conduct research, said that the purpose would be "confirmatory," whatever that meant. It has never been quite clear what was to be confirmed. If it was to confirm the safety of reactors designed and built in accordance with the NRC's criteria and regulations, then I would have to say it has been quite successful. So far, things have been pretty safe.

If we try to talk about safety in terms of risk, we are now talking about calculated consequences and calculated probabilities when we get into this probabilistic risk assessment, which is a most interesting and powerful tool but one which leads to endless arguments. Any reduction in risk can only be a claimed reduction in risk, because obviously it has not happened yet.

If improved safety is a goal, and it has not been a specific goal, it is still difficult to evaluate. Improved safety was brought in as something different from what they are doing. As I said, the safety record has been good. Reduction in risk we cannot prove. We think reduction in risk should really be the objective of the program, and we have urged that reduction in risk is probably one of the best ways of setting priorities for the program. We keep telling the staff that and they keep telling us it is difficult. It is not quite clear whether it is the mechanics of it that is difficult or the fact that the answers that come out frequently differ quite a bit from those that the licensing user-need offices want answered.

Although it probably has never been stated explicitly, a really important function of the research program has been to improve the staff's knowledge and understanding, to make the staff smarter, to make them do their job better. I think it has been quite successful in this. I think the result has probably been a reduction in risk.

Another result of the research program that nobody has mentioned, although it certainly deserves emphasis, is that it provides the NRC staff—and, I will have to admit, the ACRS—with a reservoir of knowledgeable and experienced people, mostly in the national laboratories, who can provide independent and unbiased advice and technical assistance on safety-related matters when they are needed. It is nice to have people who have done research with a regulatory bias and have them available when you have a problem to help you solve it. This was one of our problems in the early days with AEC, when all of the people were working for Mr. Shaw, in the Division of Reactor Development and Technology. It was awfully hard to find an expert anywhere who was not involved in promotion and development and could help you ask the hard safety questions. They might know them, but they were not in a position to help you. I think this has been an important but not a very obvious product of the research program.

That concludes my remarks. I will be happy to try to answer any questions on what I have said, or what was in the written testimony, or what was in our reports to the Congress.

[The prepared statement of Dr. Siess follows:]

TESTIMONY OF

DR. CHESTER P. SIESS, CHAIRMAN
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON THE SAFETY RESEARCH PROGRAM

PRESENTED TO

SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION
COMMITTEE ON SCIENCE AND TECHNOLOGY
U.S. HOUSE OF REPRESENTATIVES

May 18, 1982

NOTE.—Accompanying this statement is NUREG 0864 entitled "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1983—A Report to the Congress of the United States of America." This report of the Advisory Committee on Reactor Safeguards can be obtained from the National Technical Information Service, Springfield, Va. 22161.

I am pleased to present ACRS views in this testimony regarding the areas of interest you identified in your letter of April 30, 1982. I will also try to answer any questions you may have with respect to the NRC Safety Research Program, although the views I express may of necessity be my personal opinion in any areas where the Committee has not expressed a collegial view.

Mechanism for establishing priorities for program components

Because of the confirmatory nature of the NRC research program, priorities for NRC research activities have been based for the most part on the degree of urgency or interest expressed by the various user offices within NRC. The user office is normally the licensing branch or division that will make use of the information directly in its licensing, or its support of licensing activities. The risk reduction potential of particular elements of the research program have therefore had considerable variation depending on the degree of concern and urgency assigned by the particular user and reflected in the budget request submitted to Congress. Commission guidance has generally come primarily during the budget review process.

The ACRS was encouraged to see the use of a mechanism in the NRC formulation, during 1978, of a program to improve reactor safety that took into account the risk reduction potential of specific areas, and in its March 13, 1978 report, the Committee endorsed a proposed program (ACRS report NUREG-0438, Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants) for the improvement of methodology for evaluating research topics. In its July 1979 report (NUREG-0603, Comments on the NRC Safety Research Program Budget) the ACRS suggested that value-impact methodology could be used in determining the potential value of research programs considered to be primarily confirmatory in nature as well as the value and impact of proposed new criteria. At the same time, the ACRS expressed its belief that such methodology has only marginal value for the selection of new topics relating to improved reactor safety. The ACRS endorsed the procedures followed in NUREG-0438 for consideration of risk-reduction potential as quite adequate for this latter purpose, particularly since all the proposed TMI-related research to improve reactor safety fell well within the scope of the five areas identified in NUREG-0438.

The need for a refocusing of the NRC research program was reinforced in the ACRS report (NUREG-0603) as a result of implications of the TMI-2 accident which identified the need for work on anomalous transients and small-break LOCAs; molten-core retention; steam explosions; and accident studies, including the course of events and possible potential consequences of a broad spectrum of accidents well beyond the then current design basis accidents.

Continued concern in this area was expressed in last year's ACRS report to the Congress (NUREG-0751 dated February 1981) in which the ACRS noted that greater emphasis in the program should be placed on studies of serious accidents and on studies of plant operations and systems behavior. Risk assessment and research to improve reactor safety were also decision units singled out for substantial increases in funding. The ACRS recommended an increase in these latter items even if it should require a concomitant reduction in other portions of the overall safety research program.

Specifically, the ACRS recommended in NUREG-0751 that:

". the Commission should provide prompt policy guidance with respect to major open safety issues

. the NRC staff research user offices would have to reevaluate their approach to formulating requests for research and strive to consider these in some broad framework which takes into account the major issues confronting the agency

. the Office of Research of NRC would have to reevaluate its current and proposed programs in terms of risk reduction potential and major regulatory needs

. the NRC will have to judge whether some research, particularly that which involves large scale component testing or the application of existing methodology, should be the responsibility of the industry or DOE rather than of the NRC

. the NRC should reduce sharply that research which is merely confirmatory in nature where there is good reason to believe that current regulatory requirements provide adequate protection to the public; the research program should be geared to providing that information which is most important to NRC decisionmaking on degraded core and core melt accidents as expeditiously as possible."

In an ACRS letter to Senator Alan K. Simpson last year (June 10, 1981) the ACRS noted that it would be helpful in establishing research priorities if the Commissioners would address questions such as:

". How should resources be allocated among safety of (or risk from) plants now operating, plants already designed but not yet reviewed for construction permits (not yet built), and plants not yet designed?

. How should resources be allocated between research on accident prevention and accident mitigation?

. How should resources be allocated between research to reduce real risk and research to reduce perceived risk, if these should be different?

. How should resources be allocated between research to convince the NRC Staff that a plant is "safe" and research to convince the Atomic Safety and Licensing Board or the public that a plant is "safe?"

. When should research be done by NRC, when by the Department of Energy (DOE), when by industry, and when by a combination of these?"

The need for increased emphasis in several areas associated with the research needed to support NRC's Severe Accident Rulemaking was expressed most recently in the Committee's February 1982 report to Congress regarding the FY-83 Safety Research Program (NUREG-0864, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program for Fiscal Year 1983).

Some additional progress has been made with regard to a mechanism for Commission guidance of the NRC research program by the promulgation of a Policy and Planning Guidance Document (NUREG-0885, Issue 1, U.S. Nuclear Regulatory Commission Policy and Planning Guidance 1982) which addresses the research program as well as other regulatory matters, and the recently implemented NRC efforts to develop a long-range research program plan. The Long-Range Plan for example represents major progress toward a better coordination of NRC research planning with the budgeting process and high-level interaction between the Office of Research and the research user offices. Both of these efforts suffer, however, from the confirmatory nature of the NRC research program. The ACRS believes that the distinction that has been made between "research to improve reactor safety" and "confirmatory research" is no longer useful. Inasmuch as the purpose of the NRC is to ensure safe design, construction and operation of nuclear power plants, it should follow that any research is legitimate that contributes to this purpose without compromising the credibility or independence of the NRC. This matter is addressed in the recent ACRS report of April 5, 1982 on the NRC Long-Range Research Plan for FY 1984-1988 in which the Committee notes that:

> "Although progress has been made in better defining the objectives of the various research programs in terms of questions to be answered, much still remains to be done in identifying and assigning priorities to those problems that represent the greatest potential contributors to risk.
>
> "We repeat our previous comment that the Plan is not a true long-range plan but only a projection of current programs and programs planned to answer current questions. There has been little or no effort to anticipate future questions. For example, the Plan does not address research on LMFBRs or other advanced-reactor types beyond the CRBR. Although the scope of the Plan may be consistent with the Commission's desires and directives, it seems inappropriate to call it a Long-Range Plan.
>
> "We hope to continue discussions with the RES Staff, and perhaps also with the Commission, regarding the purpose, philosophy, scope and effectiveness of a Long-Range Research Plan."

Relationship of NRC research to the regulatory process

The NRC research program is primarily confirmatory in nature in direct support of the Commission's licensing activities and the development of related regulations. The recent integration of the Office of Standards

Development and the Office of Nuclear Regulatory Research has given increased emphasis to the coordination of research related to the development of standards, particularly those dealing with severely degraded cores. In its report to the Congress regarding the proposed FY-1983 NRC research budget (NUREG-0864 dated February 1982), the ACRS noted that, although some progress has been made, additional shifts in emphasis are still needed with respect to information about causes and likelihood of accident initiators; the important phenomena involved in accidents; and alternate design and operational features for plants in operation or under construction and for plants yet to be designed. Some specific areas warranting higher priority were identified as:

- Human Factors, including:
  - Improved bases for decisionmaking in the control room during complex transients,
  - Development of an improved approach for judging the effectiveness of licensee management,
  - Improved understanding of the impact of maintenance and testing activities on plant safety.
- Control Systems and Plant Operational safety: Higher priority and greater funding should be given to the role of control systems in safety.
- Possible design approaches to help protect against sabotage by an insider: Work is needed to supplement that effort already given to access control.
- Aging of equipment and systems: Added knowledge is needed regarding the effectiveness of surveillance programs, taking into account the degradation of nuclear plants with operating use and exposure to operating temperatures and pressures as well as the impact of radiation exposure.
- Quality in Design and Construction of Nuclear Facilities: Construction and operating experience continue to highlight the need for added emphasis on improved quality in LWRs.
- Standardization: The impact of standardization on the safety and reliability of nuclear plants needs better definition.
- Decay Heat Removal: Improved approaches to more reliable heat-removal equipment for both the reactors and the containment need consideration.
- Severe Water Hammer: Expanded knowledge about the likelihood and effects of severe water hammer during a wide range of possible transients and accidents is needed.

Relationship of NRC research to nuclear safety research at DOE and in industry

As noted above the NRC program has been primarily confirmatory with respect to regulatory activities and is for the most part independent of industry and DOE programs. The ACRS has recently endorsed the proposed plan by DOE (ACRS Report on the Final Draft of the Department of Energy's Response to Public Law 96-567 dated December 15, 1981) to establish a number of working groups with representatives from industry, NRC and DOE to identify areas needing research and to develop national programs in each area and recommend measures for their implementation. The ACRS believes that this approach is an acceptable way to develop meaningful programs with appropriate participation by the various organization for the development and execution of a research, development and demonstration program for improving the safety of nuclear reactors. Until such a coordinated effort is established, however, the ACRS has recommended in its report of February 1981 (NUREG-0751) that several areas including areas such as fire replication testing, structural engineering testing, large scale components testing or application of existing methodology, and waste management should be the responsibility of industry or DOE rather than that of NRC. In its more recent report of July 1981 to NRC (NUREG-0795), the ACRS noted the need for policy guidance from the Commission regarding the areas which can and should be the responsibility of industry.

Correspondence of funding levels to relative risk and known incidents in actual operating experience

NRC has made considerable progress in several areas, particularly those regarding reliability of existing plants, including emphasis on a more detailed understanding of plant performance which might prevent many operating occurrences which have potentially serious implications, studies of degraded cores resulting from TMI-2 experience, and studies of the physical and chemical behavior of fission products in post-accident environments. Some effort has been initiated regarding the role of control systems in safety; plant operational safety, including system behavior as a function of design; and an approach to supplement or replace the single failure criteria, all of which have been given increased recognition as a result of nuclear plant operating experience and analysis of one form or another.

An effort to examine alternate decay heat removal systems has also begun as a result of increased recognition of the vital role of decay heat removal under non-LOCA conditions. The ACRS has indicated, however, in its report of February 1982 (NUREG-0864), that even greater emphasis is needed in several areas noted above.

Impact of NRC safety research on actual safety in commercial power plants

The most significant research findings to date have not led to new regulations but have been truly confirmatory in that they have confirmed the conservatism of the NRC requirements for safety standards. Many of the loss-of-coolant accident/emergency core cooling system (LOCA/ECCS) findings are of this type. Another significant portion of the research effort has led to increased understanding of the phenomena and systems behavior important to safety. This allows the NRC Staff to better judge the safety of proposed designs or the conservatisms of existing designs and practices and also provides the background needed to handle unanticipated problems when they arise. Only a portion leads directly to new criteria for new regulations. Examples of such contributions are:

- The initial safety research effort was primarily to examine the adequacy of the NRC LOCA/ECCS evaluation models which had been developed as part of NRC ECCS rulemaking proceedings during 1972-73. These models were used to predict systems and core behavior during LOCA/ECCS transients. The related test program has demonstrated the overall conservatism of the NRC Appendix K evaluation models; contributed to a better understanding of the phenomena involved; contributed to the development of best estimate codes for improved prediction of such transients; and identified a number of specific deficiencies and areas of overconservatism in the modeling of the phenomena involved. More specifically, information from NRC-sponsored research has indicated that it may be appropriate to amend Appendix K of 10 CFR 50 to reduce the rate of heat generated from radioactive decay of fission products and zirconium-water reaction that is used in the ECCS model, and changes in the clad rupture model. The NRC Staff has proposed such amendments. The ACRS concluded, however, that it might not be prudent to reduce the existing safety margins piecemeal. As a result, the NRC Staff had deferred action on these amendments and undertaken a comprehensive reevaluation of the overall margin in the ECCS model. This effort was interrupted by the TMI-2 accident and apparently is now of low priority because of other licensing activities. An additional benefit is that the best estimate codes developed as a result of this work will contribute to the capability of reactor training simulators to model accident transients.

- Although the ACRS has recommended termination of the LOFT program by the end of FY-82, it was not because we believe that LOFT has not been useful. To the contrary, we believe that LOFT has accomplished its early goals and that further tests are not as cost effective as some research that we have identified to be of higher priority. We believe that LOFT has provided a better understanding of both the large and small break loss-of-coolant accidents. LOFT has also provided valuable insights in resolving the so-called hot wall effect; has demonstrated that natural circulation cooling and refluxing are viable methods of heat removal; has demonstrated that recirculation pump current is an indication of system conditions; has simulated an accident where a steam bubble formed in the pressure vessel and has demonstrated that available computer codes can predict it. LOFT has also been used to study the desirability of requiring that the main coolant pumps be turned off or left on during a small break LOCA. These results have had a significant impact on licensing decisions made by the NRC Staff on a case-by-case basis.

- The ORNL Heavy Section Steel Technology (HSST) Program has provided evidence to support the presumption of a leak-before-break failure mode in reactor pressure vessels. This program has also generated substantial data that have been used in predicting irradiation embrittlement and margins to failure of reactor pressure vessels when flaws are present. It has also provided input to NRC requirements for fracture toughness of pressure vessel materials as described in Appendix G to 10 CFR 50; has contributed data used in the ASME code that was made part of Appendix G; and has provided the data base for Regulatory Guide 1.99 concerning the effect of impurities on the sensitivity of steel to irradiation.

- Recent results from research on steam explosions has demonstrated a much smaller energy release than was previously assumed in the Nuclear Safety Study (WASH-1400). This is expected to have a significant impact on the probabilistic risk assessment of core melts and consequently on licensing requirements related to Class 9 accidents.

In addition to these specific applications, it is well to remember that one can never predict future problems with any certainty. Thus, one must try to anticipate regulatory questions to provide fundamental information and a small cadre of knowledgeable people to deal with safety problems when they arise. A few examples are:

- Work on LOCAs due to the rupture of a large primary coolant pipe helped NRC to quickly obtain answers to questions concerning small break LOCAs and cooling by natural circulation after the accident at TMI-2 raised such questions.

- The expertise resulting from the HSST program's work on thermal stresses in pressure vessels following a LOCA has allowed the NRC to move quickly on recent questions concerning pressurized thermal shock in older pressure vessels.

- An active research program on developing and applying the techniques of probabilistic risk assessment to reactor safety questions is providing an essential capability to perform reliability and risk assessments and to establish quantitative safety goals.

- On the negative side the NRC has not had an active research program in water chemistry and metal corrosion, leaving this to industry. We believe this will make it more difficult for NRC to move quickly and authoritatively in areas such as steam generator tube degradation and corrosion problems associated with long-term isolation of high-level wastes from groundwater.

UNITED STATES
NUCLEAR REGULATORY COMMISSI
ADVISORY COMMITTEE ON REACTOR SAFEG
WASHINGTON, D. C. 20555

October 1, 1981

CHESTER P. SIESS

Dr. Chester P. Siess was born in Alexandria, Louisiana, on July 28, 1916. He received a Bachelor of Science degree from Louisiana State University in 1936 and M.S. and Ph.D degrees in Civil Engineering from the University of Illinois in 1939 and 1948.

Dr. Siess is Professor Emeritus of Civil Engineering at the University of Illinois in Urbana-Champaign where he served as a member of the faculty since 1941. During this period, Dr. Siess has served as a consultant to many government agencies, architect-engineer firms, and industrial organizations on special engineering problems. His particular field of interest is the design and construction of bridges, buildings and other structures of reinforced and prestressed concrete. A considerable part of his activity at the University of Illinois has been devoted to research. He is author or co-author of numerous publications, including articles on design, testing, and performance of beams and columns and is a Professional Engineer in the state of Illinois.

He has been a member or chairman of numerous committees of the American Concrete Institute, the American Society of Civil Engineers, and other societies and has been active in the preparation of structural design specifications and codes for both reinforced and prestressed concrete structures. He is a member of the National Academy of Engineering

Dr. Siess has been awarded the Wason Medal and Turner Medal of the American Concrete Institute and was made an Honorary Member of that organization in 1969. He served as Vice-President of the Institute from April 1973 to April 1974 and as President from April 1974 to April 1975. As past president, he continued to serve on the Board of Directors until 1978. Dr. Siess has also received the Research Prize, the Howard Award and the Reese Award of the American Society of Civil Engineers.

In 1968, Dr. Siess was appointed to the statutory Advisory Committee on Reactor Safeguards and was reappointed for additional four-year terms in 1972, 1976, and 1980. He was also elected Chairman of the Committee in 1972.

Dr. Siess resides with his wife, the former Helen [illegible]nson, in Champaign, Illinois.

Mrs. Bouquard. I certainly appreciate your testimony. Not only do I appreciate your appearance and your testimony today, but I appreciate what is very obvious, the depth of your knowledge and your commitment to the NRC safety research programs. We do appreciate your being with us today.

I am going to confine my remarks to your written testimony which I would like for you to elaborate on, because I think the remarks in your oral testimony were certainly well put. You refer to the risk reduction potential. Do you feel that risk-reduction techniques would lead to a more balanced program? I understood from your remarks that you felt it was rather fragmented.

Dr. Siess. We think that some way of looking at priorities, in terms of risk reduction, clearly will be more balanced. This was part of our argument, I think, for reducing the LOFT program, where the program had already succeeded in reducing risks about as far as we thought it could go, and there were other things we thought were more important.

I do not believe risk reduction necessarily has to involve detailed, complicated, probabilistic risk analyses. When the staff came up with five topics in NUREG-0438, in a program for research to improve safety, they did not do a probabilistic risk analysis. They simply took the opinions of about a dozen people, including the ACRS. They looked at the number of reactors that could be affected and the kinds of things that could be prevented. And on a qualitative, somewhat subjective but very knowledgeable basis came up with five items for research.

Mrs. Bouquard. Do you feel they have a lot of research for its own sake?

Dr. Siess. These were all good, hot items. Three of them I think ended up high on the list after TMI, and the ranking was done before TMI. Here was a good attempt to come up with items that looked like they would be important to reduce risk, but it was not extremely quantitative and formalized. If we have good quantitative PRA, use it. But our job is to reduce risk; that is what NRC exists for. It seems to me that is the best way to set priorities.

Mrs. Bouquard. What I was trying to get at is, do you feel that a lot of this research has not been applied? The research is good and it has been done, but it has not been put into direct application?

Dr. Siess. I can think of some things that have not been applied. There are some results that have come out of the LOCA/ECCS program, and I think Mr. Minogue mentioned these, the decay-heat curve and the clad oxidation criteria. The staff came in and said, "Look, we have learned this. Let's change appendix K." The ACRS said, "No. Let's not do things piecemeal. Let's wait." But there are other things that have come out. Some have been a relaxation. There are some places where it has tightened up. It is always easier to tighten up the regulations as a result of new knowledge than it is to relax them, because you do not need the same level of confidence to tighten up as you do to relax, not if you are sitting on the regulatory side.

Mrs. Bouquard. Do you have any other examples you would like to cite?

Dr. Siess. In our written testimony, we listed a number of places where we felt the research had been useful. One good example, I think, is the heavy section steel technology program, which has been going on for a great many years. Not only did it address problems that we knew existed, but it provided information to address problems and provide answers to questions that we are now asking about new problems: The so-called pressurized thermal shock problem. We would be dead in the water trying to handle that problem if we did not have all the information on fracture mechanics from the HSST program. That, of course, is beautiful research. That is getting the answers before you need them. But the NRC program really is not set up to do that. It really cannot be set up to do it 100 percent, or 90 percent because they have too many questions day to day that they have to answer, the user-need type of thing, and confirming judgments that have already been made.

Mrs. Bouquard. In your report in February, you said you did not feel that the program was particularly well focused. Do you have an approach that you think would lead to better focus, where we could spend our money more efficiently?

Dr. Siess. Well, risk reduction would certainly help. It is not easy. We have to review a program that involves reactor safety including light-water reactors. There are some elements relating to the Clinch River breeder reactor. There are some elements relating to waste management. It is awfully difficult to say that some aspect of high-level waste management is more or less important than, say, pressurized thermal shock in light-water reactors. So there are areas where you would have difficulty setting priorities from one to the other. It is always done, but it is not easy.

Within the reactor safety area risk reduction is clearly a good basis for focusing it. User needs are a good basis for focusing a short-term program in areas where somebody has need for information. It does not mean that it will be the best result in the long run, because they might not have asked the right question. The severe accident research program which is part of the severe accident phenomena mitigation effort, cuts across several areas. This is a tough one, and that is one we did comment on that I think was poorly focused. The answers they were going to get were going to take 5 years, and the licensing people needed them in two.

If you set up a program to give users the answers they want in 2 years, they may not have asked the best questions, and you may have to start over again. I do not know. Some of the results are not going to work out when you are trying to get answers quickly. But risk reduction is a good way of focusing it. It is not always used; people who yell the loudest get the answers.

Mrs. Bouquard. Turning to human factors research, the ACRS recommended that the NRC define a broad area where industry could take a lead and accept responsibilities. Just where do you think industry could take a lead in an appropriate manner?

Dr. Siess. This is one area where I will plead inexpertise, but in our report to Congress last February we did mention some areas where we think INPO is working. I think they are doing a great deal of work. Industry, both the utilities and the nuclear steam supply vendors, are working hard on improving procedures for control of the plant during normal operation and during abnormal conditions. The safety parameter display system is being developed. Although it is required by NRC, there is a tremendous amount of work being done by industry on that, going far beyond what some people had in mind.

If we get into the areas of maintenance, which do concern us because we see an awful lot of abnormal situations produced by either a maintenance or a testing error, there I think we have to rely tremendously on the industry.

Mrs. **Bouquard.** You feel that industry has an essential role in this area, in maintenance and testing activities and plant safety?

Dr. **Siess.** I think so. Not exclusively industry, but they are going to have to get a lot of help.

Mrs. **Bouquard.** They will have a great impact on decisions. Would you explain your position on NRC grants to universities on safety research?

Dr. **Siess.** That one is a little bit complicated, and some of us are rather hesitant to put it into our recommendations, since there are so many university professors on the committee. However, a number of them are like me, retired and no longer concerned.

NRC has a problem—I am not sure it is limited to NRC. They have a limited authorization for grants, but they were told somewhere that the grant has to be something that the university wants to do, and it should not be something that NRC wants them to do. In other words, there was a feeling that if it really was going to be useful to the NRC, the university should not do it. I think that is a distortion however.

Most of NRC's research goes to the national laboratories. For a great deal of it, they are the only possible place. The national laboratories, of course, will not submit a proposal in response to a request for proposal. If you want them to do it, you go and ask them. They will not compete. That, presumably, is DOE's order. If you send out a request for proposals, then the university must compete, and it becomes very complicated. A lot of universities do not want to compete. If you try to do it without a request for proposals, it then becomes a sole-source contract, and it takes more paperwork than most people are willing to get into. You have to establish that they are the only people who can do it, and so on.

So very little is put out to universities, and yet we feel that there should be a modest amount—a little more than they are doing now—simply because it will help develop another group of experts. It will get some broader understanding of the problems, get some new people thinking about it, who might come up with some new ideas. And besides, it helps train some students that the NRC is going to have to hire one of these days, or someone else in the industry is going to have. They have to come from somewhere.

Mrs. **Bouquard.** That point is well taken.

Dr. **Siess.** It is not a big program. In fact, it got cut to zero, I think, with fund cuts. We just thought it deserved mention. I believe we mentioned it also in our first report to the Congress 5 years ago.

Mrs. **Bouquard.** You feel there should be a mix with the universities and the laboratories as well for a balanced approach?

Dr. SIESS. Universities will not be competing with the laboratories for multimillion dollar contracts; they will be doing small jobs. That is all they can do. Incidentally, they do them a lot cheaper. Graduate students come cheaply, even today.

Mrs. BOUQUARD. I believe ACRS believes strongly that the NRC ought to be paying more attention to research on advanced reactor study, particularly post-CRBR LMFBR's. Would you explain your thinking on this?

Dr. SIESS. I will try. You are fortunate that Dr. Corbon is not here. I know a lot less about it, so I can probably do it much faster. He could give you an hour lecture on it, I am sure.

Mrs. BOUQUARD. We are looking forward to your version.

Dr. SIESS. We have been building light-water reactors commercially for 25 years or in that neighborhood, and we are still doing an awful lot of research. A lot of it we probably could have done and should have done before we got into this. It is just a feeling here that if we are going to go ahead with a breeder reactor program after CRBR, there are going to be some differences between a large commercial breeder and CRBR. It is time to start asking questions, trying to decide what are the important questions, safety-related questions, and getting some research started now before people start designing these things and going through the same kind of process we did on light-water reactors. There is no reason to think we are so much smarter that we know how well we should build them and all the things we should think of. But we feel it is worth an attempt to try and get started on the research program somewhat in advance. We have never been happy that all we can look at now is the specifics of Clinch River, and nobody is looking further ahead.

Mrs. BOUQUARD. You expressed some concern about the goals of the German UPT facility and their priority in the research program. How would you recommend they redirect their portion of those expenditures?

Dr. SIESS. This, again, I am not familiar with. If it turns out that the answer I give you is wrong, we will let you know. As I recall, our concern was that the German upper plenum test facility is really aimed at answering some questions they have about their systems, and their systems differ from our systems. They inject emergency-core cooling water into both the hot and cold legs, and we inject only into the cold leg. This is a significant difference, and we felt that putting much money into that to help them solve their problem was not really going to help us solve our problem.

Mrs. BOUQUARD. I would like now to defer to my colleague from California, Mr. Lowery, a very valuable member of this committee. We welcome you and look forward to your input into our hearings today.

Mr. LOWERY. Thank you, Madam Chairman. Possibly I should call you Commodore Bouquard, because of our tour yesterday of the nuclear-powered submarine.

Mrs. BOUQUARD. That point is well taken. I think you also shared in the experience and learned a lot, including how to go up and down submarine ladders.

Mr. **Lowery**. We call that place down there in Norfolk the San Diego Naval Station—East. It is the largest naval facility in the entire free world.

Madam Chairman, I would commend you for calling these first oversight hearings on the NRC research and development programs the committee has had. I ask unanimous consent to have a statement in the record of this hearing.

Mrs. **Bouquard**. Without objection, your statement will be included in the record.

[Material to be supplied follows:]

OPENING STATEMENT OF HON. BILL LOWERY, ERP HEARING ON NRC RESEARCH AND DEVELOPMENT

I want to thank the chairman for calling these hearings, the first oversight hearings of NRC research and development programs that this committee has had. The committee has recently received partial jurisdiction for these programs, and I feel we can bring to the Agency a fresh evaluation of their activities. This committee is well experienced in evaluating research and development programs in other agencies, and we should be able to apply this experience to this new case.

In fact, we find that there are numerous differences between the NRC research program, and others that we have examined. I look forward to these hearings as being a vehicle for determining the reasons for these differences and what can be done to improve the NRC research program.

In the last Congress, we passed the "Nuclear Safety Research, Development, and Demonstration Act of 1980," (Public Law 96-567) which established the Department of Energy as the lead agency for Light Water Reactor (LWR) safety research, and directed them to prepare a comprehensive program management plan in consultation with NRC and others. I hope that this hearing will be able to add greatly to our understanding of what DOE should be doing, what industry should be doing, and what the NRC's program should be. In contradiction to the act, however, the present policy is to give the responsibility for LWR safety research to NRC; DOE requested only $1 million for fiscal year 1983, while NRC has an authorized program of $230 million.

Finally, the primary result of NRC research and regulation should be to improve the safety of the public, while providing for improvement in the ability of the Nation to utilize the benefits of nuclear power. NRC has recently issued its proposed safety goals related to reactor accidents; these goals are still out for comment. However, I applaud the efforts of the Commission in seeking to put the risks of nuclear power in context, and at the same time also answer the question, "How safe is safe enough?" In the Nation today, the rate of death from accident is 6 out of every 10,000 persons. The NRC goal would result in only a 0.1 percent increase in the risk to the population resulting from a nuclear accident. The risk of death from cancer in the Nation is two out of a thousand; the NRC goal would limit the increased risk to only 0.1 percent. While no death is ever "acceptable," a risk which is three orders of magnitude lower than the presently accepted risk, is certainly not too great an increased burden.

Thank you, Madam Chairman for calling these witnesses; I look forward to hearing from them today and tomorrow.

Mr. **Lowery**. Second, I ask unanimous consent to submit some written questions, both to Dr. Siess as well as to Mr. Minogue.

Mrs. **Bouquard**. Without objection.

Mr. **Lowery**. Dr. Siess, I do have a specific question for you. Being from California, I will take advantage of your expertise as a structural engineer concerning seismic design. A less publicized episode occurred in San Diego County at the nuclear facilities being built at San Onofre. Although the stainless steel vessel was installed in the opposite direction, this did not matter because it was a mirrored image. But there was a problem with a piping hookup that resulted in an issue about the advisability of a stiff structure versus a more flexible structure for surviving earthquakes. Would you comment on this from a seismic standpoint, as well as on a far more publicized controversy at Diablo Canyon?

Dr. SIESS. The question of stiffness versus flexibility it is extremely complicated, of course; but in general, the more flexible things are, the better they act during an earthquake. We have tended in the nuclear business to go in the opposite direction—to restrain everything—to tie every pipe to the building so that it moves with the building and shakes with it. This has bothered us.

The way we go about the analysis and the way we go about the design, we are forced into it. It is not too clear just how we can relax it, because it gets involved in inelastic behavior, which is much more difficult to analyze. The analysts who work with computer programs would much rather deal with elastic conditions; they are much more amenable to the calculation. So we are stuck.

We are in an area where more is not necessarily better. Making it stronger tends to make it stiffer, which tends to make it subject to greater forces in an earthquake; it goes in the wrong direction. But we have no well-established design procedure to design for flexibility for nuclear plants. We do it for structures, but we have had a lot of experience. We have a lot more buildings subjected to large earthquakes than we have nuclear plants. So that problem is essentially unresolved.

Mr. Minogue mentioned the directions the staff is looking at now in terms of research, an add on, in addition to the Seismic Safety Margins Research Program (SSMRP), and I think it might eventually come out with something. But I cannot help you very much. We have had people tell us that they have looked at conventional power plants after major earthquakes—there was one near Caracas, Venezuela, a few years ago; there was one in southern California, the last El Centro earthquake—and they did just beautifully. No pipes failed, and so on. They say, "Well, a nuclear plant ought to do better." But I am not sure it should do better. It is designed much more rigidly—more pipe restraints, more snubbers. So I am not quite sure of that parallel: that it would do better because the design is much more rigid.

Regarding Diablo Canyon, I wish I could answer you. I am chairman of the subcommittee of the ACRS for Diablo Canyon. The members keep asking me, "When are you going to have a meeting and let us review it?" I say, "I am going to wait until I find out what happened." I am getting every piece of paper that comes out on Diablo Canyon, and they are just barely beginning to identify deficiencies in the design. They have identified a great many deficiencies in the quality assurance program, but they have not found too many deficiencies in what was built.

Mr. LOWERY. "They" being who?

Dr. SIESS. The people who are reviewing it. Right now, that is Cloud, and Teledyne is in on it now. I have not seen anything out of Teledyne, but I have been getting Cloud's so-called errors and omissions reports, and he finds that a pipehanger was 6 feet from the end of a pipe instead of 5 feet, as analyzed. But until somebody has gone through the calculations to see what difference this makes, we do not really know what the significance is.

There will be some deficiencies. I do not know how significant they will be. There have been cases where they have found stresses that were 100 percent higher than they were initially computed, but in one case they were 10 percent of the allowable stress and in another, 20 percent. So we do not really know yet. They are still getting into the deficiencies in the design rather than in the QA program.

Mr. **Lowery**. Thank you very much. Thank you, Madam Chairman.

Mrs. **Bouquard**. Thank you very much, Mr. Lowery.

Dr. Siess, we thank you for being our witness today. We will be submitting questions to you which we would like to include in the record of hearings today, and we hope you will be so kind as to respond to us.

Dr. **Siess**. We will do our best.

Mrs. **Bouquard**. Thank you very much.

Our witnesses have been so very fine today. We appreciate the work that you have put into your testimony, as well as your knowledge in answering the questions before you.

The subcommittee will continue its hearing on the NRC research program tomorrow afternoon at 1:30 in this same room 2318. The committee is now adjourned.

[Whereupon, at 3:16 p.m., the subcommittee recessed, to reconvene at 1:30 p.m., Wednesday, May 19, 1982.]

OVERSIGHT

THE NUCLEAR REGULATORY COMMISSION RESEARCH PROGRAM

WEDNESDAY, MAY 19, 1982

House of Representatives,
Committee on Science and Technology,
Subcommittee on Energy Research and Production,
Washington, D.C.

The subcommittee met, pursuant to notice, at 1:34 p.m., in room 2318, Rayburn House Office Building, Hon. Marilyn L. Bouquard (chairwoman of the subcommittee) presiding.

Present: Representative Marilyn L. Bouquard.

Mrs. Bouquard. The subcommittee will come to order.

Good afternoon, ladies and gentlemen. Today we continue our hearings on the research program conducted by the Nuclear Regulatory Commission.

We will hear first from the Department of Energy which conducts a number of research programs intimately related to those sponsored by the NRC.

We are particularly anxious to hear how the Department's safety research program, mandated by the Nuclear Research, Development, and Demonstration Act of 1980, will interface with NRC research activities.

We are also fortunate to have with us today representatives of two outstanding industry groups. John Taylor from the Electric Power Research Institute will provide an industry perspective on the focus and priorities of the NRC research program. Dr. Edwin Zebroski, from the Institute of Nuclear Power Operators, will address his remarks specifically to the impact of plant maintenance and operations requirements on nuclear reactor safety research.

We will also hear from Dr. Herbert Kouts who will represent the views of the Nuclear Safety Oversight Committee's Reactor Safety Research Review Group. We understand that his testimony was developed jointly with Dr. Norman Rasmussen, who will not be able to join us today because of scheduling difficulties.

I realize unanimous consent that pursuant to House rules, television and other recordings be permitted at today's hearings.

We are happy to have Gordon with us today. We appreciate his work, working with the committee and now with DOE.

Gordon, if you would like to proceed with your testimony, you may at this time. We will place a copy of your written testimony into the record and you may summarize if you wish.

STATEMENT OF GORDON L. CHIPMAN, JR., DEPUTY ASSISTANT SECRETARY FOR NUCLEAR REACTOR PROGRAMS, U.S. DEPARTMENT OF ENERGY

Mr. CHIPMAN. Thank you, Madam Chairman.

It is a real pleasure to be here. I appreciate the chance to come before you today and talk about safety research programs. As we both know, it is a very important project and a concern of the whole Nation.

My comments today are primarily derived from the Department's interactions with NRC and the nuclear industry which includes National Laboratories, Department contractors, and universities as we have tried to fulfill our obligations under Public Law 96–567, the Nuclear Safety Research, Development, and Demonstration Act of 1980 which of course came out of this subcommittee. It is this act and the Department's efforts to respond to it that form the basis for my comments this morning.

Rather than go through the complete text of my testimony, I would like to shorten it somewhat since the full text has been made available for the record.

The Safety Act takes the broad view that safety improvements are to be considered in light of their costs, their benefits and that in addition to these two factors, which of course are very important, complexity should be minimized in the process. Complexity alone can be a safety concern.

The Department believes that the objectives of Public Law 96-567 cannot really be met by focusing separately on improving each of the commonly identified technology areas. Therefore, we believe that a comprehensive definition of program requirements is needed and we are developing such a definition to determine just exactly what research and development is appropriate in each technology area and what the relationships are between these technology areas and improving the overall safety of light-water reactors.

In this process we have convened and organized 10 working groups with representatives from the Department of Energy, the Nuclear Regulatory Commission, the Institute for Nuclear Power Operations, EPRI, the utilities' IDCOR program, utilities' owners groups, individual utilities, DOE contractors, the AIF, the EEI, the ANS, national laboratories and universities—a rather broad range of groups—to assist the Department in developing the program requirements.

In order to better coordinate these efforts, we are working to consolidate the existing 10 groups into no more than 4 because we think that will be beneficial.

The overall objectives of this DOE/NRC/industry working group effort is to develop a comprehensive identification of what the safety issues are and what efforts are required to resolve these issues. Also we want to review the work that is underway currently today to resolve these issues.

Third, we would like to define the programs that would resolve the remaining issues that have not been addressed.

Last, we expect these efforts to assist the Department of Energy in developing a plan for prioritizing and implementing these programs. In carrying out this activity, we plan to coordinate with NRC and to the extent necessary, to develop memoranda of understanding and other vehicles for cooperation and coordination with Government agencies, foreign governments, and industry, as appropirate.

A major output of these working groups will be identification of safety research and development areas that warrant attention and, very importantly, in order of relative priority.

Mrs. Bouquard. Gordon, excuse me. I believe I am going to have to go to the floor. We will stand in recess so that I can make this vote.

[Whereupon the subcommittee stood in recess at 1:41 p.m.]

Mrs. Bouquard. We will resume our hearing.

Mr. Chipman. Thank you, Madam Chairwoman.

I was in the process of describing the program that DOE has established to develop a coordinated safety research and development program. There are a total of 150 experts from some 58 different organizations in the working groups we have formed. The NRC has been working closely and most cooperatively with us in these efforts. We, therefore, think this is a broadly based effort, probably more so than any past effort at developing a coordinated program.

This effort should provide us—the Department, the NRC, the industry, and also the Congress—with a very useful tool to identify and satisfy LWR safety requirements.

Based on the efforts that we have undertaken and the review of NRC program plans, I believe that significant improvements can be made in existing safety research and development programs in the applicability and utilization of research and development results.

Our efforts in responding to Public Law 96-567 have led us to identify a very important goal of effecting better coordination of regulatory and operational requirements.

Historically the NRC has focused its research efforts on demonstrating or confirming the conservatisms of its regulations. Its role as an independent agency has naturally led to a degree of isolation from outside views that might challenge this narrow focus. I am afraid that consequently they have been isolated from many of the operational and potential safety implications of this approach.

As a result of the TMI accident, while the general level of safety of nuclear power plants was demonstrated and confirmed, in my view—and I think in the view of all experts in this area—improvements have been achieved in broadening the view of what is of safety significance. This committee's interest in this area is a clear demonstration of that. As a matter of fact the Safety Act itself was an outfall of the TMI accident.

It does appear, however, that further improvements in the area of safety research and development, including coordination among all interested parties and simplification of requirements, are appropriate. I would like to expand on what I mean by this.

The purpose of the safety research program should be to provide the information that is necessary to assure an acceptable level of risk to the public from the operation of nuclear power plants. In this regard, there are two important ways in which the risk of nuclear operations may be minimized.

One is to insure that the plant design and operation address the maximum range of potential accidents. The second is to improve the reliability of plant operations. A program which balances these considerations would be very effective in improving safety.

We believe the historical lack of coordination of research and development programs within the developmental, design and regulatory organizations has led to an overemphasis on the first element and insufficient attention to the second.

We are seeking through our efforts under Public Law 96-567 to improve the effectiveness of the safety research and development. We believe that efforts of all parties in the working groups that we have established have overcome many past barriers to cooperation and in fact have provided the means to develop a truly balanced safety research and development program.

One of the manifestations of the historical approach to safety research is the major emphasis that is being placed and has been placed on the development of refined analytical techniques which can calculate the effects at very low probabilities. This capability has little or no relationship to actual construction or operation needs and practices. The result, when modifications to design and operations have been made to provide what appear to be safety improvements, is that the reliability of plant operations can be negatively impacted. There is even the potential of safety implications of such actions.

We have investigated in considerable detail one such situation which we believe to be typical of the kind of problem that can result. The example has to do with the pipe constraint analysis techniques and design criteria which have been developed to provide accommodation for low probability earthquakes and other low probability piping loads. This analytical emphasis on low probability concerns has resulted in severe and detrimental impacts to operating nuclear powerplants.

During the course of the investigation into this matter it became clear that the practice of restraining piping to minimize movement during earthquakes results in a system which is not necessarily compatible with the requirements for thermal expansion and contraction during normal operation.

As a matter of fact, Licensing Event Reports, on incidents in the field, confirm that severe loads and overstress have occurred as a result. Pipe snubbers, the components that hold the pipes rigid, have been pulled right from the anchor devices and right out of the wall as a result of such thermal stress.

This is a case where we have traded away one form of uncertainty—that is our inability to calculate stress in more freely moving pipe systems—for what is a known problem of rigidity. In addition, the large number of hangers, seismic supports and pipe whip restraints have resulted in crowding of work areas. Access to piping and other systems is restricted, making it more difficult for both men and equipment to be brought to the worksite.

I am providing for the record a copy of the executive summary of our report on this matter to the committee. The complete report is also available for anybody who is interested.

Mrs. BOUQUARD. Without objection, it will be made a part of the record.

[Material referred to follows:]

REGULATORY REFORM CASE STUDIES

I. EXECUTIVE SUMMARY

This report focuses on issues associated with the President's regulatory reform program. A previous report prepared by UE&C(1) focused on the key issues to be resolved to improve the regulatory climate and streamline procedures. The report recognized the paralyzing effect of attempting to approach zero risk and to obtain legal proof of perfection. This approach leads to goals which are fundamentally unattainable and which in fact lead to counter productive efforts which can potentially contribute to reduction in safety margins.

The objectives of reform are related to the reduction of overall schedules and to the lowering of costs by reducing construction manpower, equipment costs and commodities. Consistent with these objectives is the goal to provide a more practical approach to engineering design and construction consistent with realistic safety goals.

In order to assist DOE in its responsibility to recommend improvements to the nuclear regulatory process, specific examples of regulatory requirements and their industrial interpretation were sought on a case study basis. The case studies program is aimed at exploring the commodity and labor productivity aspects of piping, electrical and concrete. This report relates to piping. Presentation on cost and productivity factors were made to the management of four utilities with total nuclear capacity in excess of 10,000 MW.

Following the management presentations, site visits were made to the facilities to discuss piping design and construction practice. Four nuclear

projects and one coal-fired power plant proje:t were studied etermine design
and construction procedures and relative productivity. The f wing points
summarize the findings:

1. On an overall basis, productivity at nuclear sites onsiderably
poorer than that on a fossil project (Average of 7. r/l.f. to
erect large bore piping at a nuclear site vs. 2.8 m f. at a
fossil plant).

2. Investigation of the reasons for lower productivity the nuclear
sites brought forth the following observations:

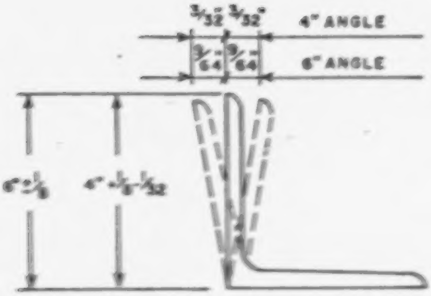
    a) The dimensional tolerances and clearances callec r on the
    piping erection, pipe hanger and piping restrair drawings
    are often in the range of 1/16" or less (in some ses, 1/32").
    As may be seen on Figure 1, these clearances are some cases
    more restrictive than the fabrication tolerances the struc-
    tural members used to fabricate the pipe support restraint.
    Many manhours are expended in erecting component these
    restrictive dimensions.

    b) Field construction forces are not accustomed to ing to these
    tight dimensions. Usual field practice is to as that piping
    fit-up to the flange or connection of a piece of ipment is
    true, and to allow the remainder of the piping t loat" on
    free swinging supports. Attempting to make ever nnection
    and support for the total piping system to be di ionally per-
    fect is time-consuming and requires an inordinat ount of
    re-work of nominally completed systems.

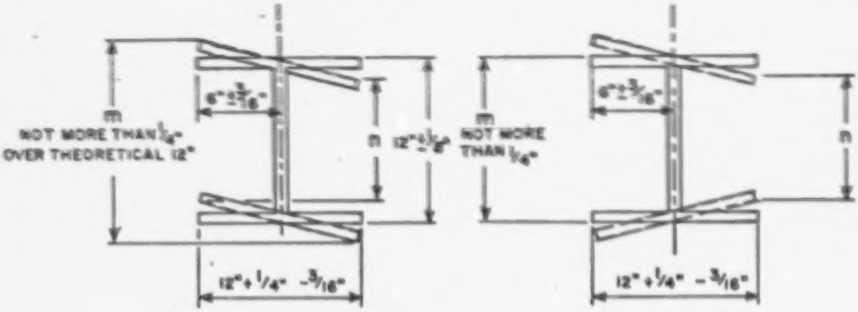
    c) The large number of hangers, seismic supports and pe whip
    restraints (all with stringent dimensional requi nts) have
    resulted in crowding areas with structural membe Access to
    piping (and other systems) is restricted, making more difficult
    for both men and equipment to be brought to the site.

    d) The construction procedures, as specified in the lity Assurance
    program of the constructor and utility owner are itored by the
    site Quality Control inspectors and by NRC for st compliance.
    There is no provision in the process for any site lated super-
    visor, Q/C inspector or other individual to exerc judgement to
    accept anything other than a "perfect" item of w ccomplished
    in compliance with established procedure. Qualif judgement,
    even based on experience and technical training, ot an accept-
    able criterion for approving any deviation.

    e) Late receipt in the field of final erection drawi has caused
    problems because of necessitating work out of seq e in order
    to keep all hands busy. In many cases, drawings revised and
    lines re-routed after they have been erected.


3/32" 3/32"
4" ANGLE
3/64 3/64
6" ANGLE
6" ± 1/8
4" + 1/8 - 1/32

DIMENSIONAL TOLERANCE ON STANDARD ANGLES


m
NOT MORE THAN 1/4"
OVER THEORETICAL 12"
6" ± 3/16"
n
12" ± 1/8"
12" + 1/4" - 3/16"
m
NOT MORE
THAN 1/4"
6" ± 3/16"
n
12" + 1/4" - 3/16"

DIMENSIONAL TOLERANCES FOR 12" WF AND SMALLER
OUT OF SQUARE OF PARALLEL $m - n = no > 1/4"$.
DIMENSIONS SHOWN FOR 12" WF.

FIGURE 1 - Standard Mill Tolerances for Structural Shapes

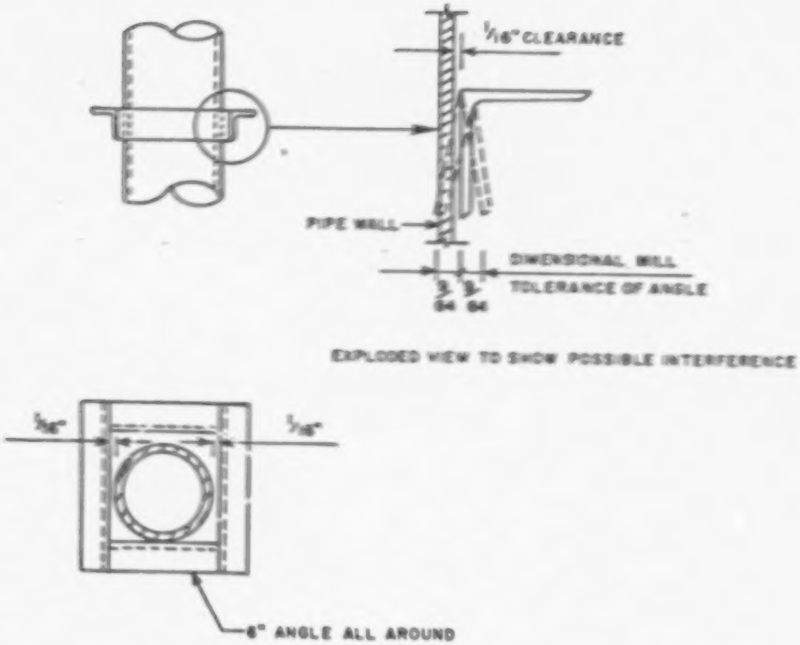

1/16" CLEARANCE
PIPE WALL
DIMENSIONAL WALL
TOLERANCE OF ANGLE
3/64
3/64
EXPLODED VIEW TO SHOW POSSIBLE INTERFERENCE
1/16"
1/16"
6" ANGLE ALL AROUND

FIGURE 2 - Typical Pipe Support

3. In continuing the case study work, it became apparent that, while the problem manifested itself in the field, a large part of the difficulty began in the licensing phases.

    a) Statements made in the PSAR and the engineer's Quality Assurance Program for each project embrace the provisions and requirements of 10CFR50 and discuss the methods of implementation.

    b) The interpretation of these requirements within the piping groups of each engineering organization has resulted in developing the dimensional clearances which are so onerous in the field.

    c) As the full impact of the implementation of the commitments made in the PSARs has been recognized, estimates for commodities and manhours have increased. At all sites visited, the increase in commodities was recognized as being only slightly affected by recent additional regulatory requirements.

    d) The dimensional mill tolerances of the structural members in some cases exceeds the dimensional clearances specified for a pipe support. When this occurs, the structural members must be machined to the proper size.

    e) Pipe stress analyses, especially when seismic movement and restraint is incorporated, result in dimensional requirements that are beyond the capabilities of conventional erection practice. There has been no discernable recognition of this by the engineering organizations, nor any guidance or direction to the constructor on how to use new techniques to surmount the problem.

4. Interviews with senior management at all plants indicated an awareness of productivity loss, but no deep understanding of underlying causes. The best attempts at alleviating the situation were to:

    a) Use detailed planning of construction activities and coordination of engineering work to support construction.

    b) Have senior engineering discipline chiefs available (at site, if necessary) to reconcile any problems as they arise.

    c) Have an informal inspection of a component or item of work to assure acceptance, and make necessary corrections prior to formal Q/C check.

    d) Have staff of Q/C inspectors with work stations close to the area where work is going on, so that waiting time by craftsmen is minimized.

During the course of these case studies, it became clear that for the initial task, piping, the practice of restraining piping to minimize movement during earthquakes results in a system which is not necessarily compatible with the requirements for thermal growth during normal operation. Each of the requirements for seismic design contained in the Regulatory Guides (RG 1.60, 1.61, 1.92 and 1.122) incorporate a degree of conservatism in order to provide for uncertainties. This results in increasing the loading input of the seismic event on plant piping systems and other components. Increasing the dynamic loads and deflections results in more rigid systems, as utilities have shifted away from the alternative use of snubbers due to poor experiences.

Many plants have been designed and partially built and then have had to go back and stiffen their systems because their dynamic loads have been revised upwards because of:

(1) A redefinition of the site specific earthquake;

(2) identification of new loads such as in the case of the BWR pressure supression containments; or

(3) the asymmetric LOCA loads of PWR plants.

There is growing concern in the industry that the current design practice of stiffening piping systems is contributing to the potential for overstress due to thermal expansion.(2) NUREG/CR-2136, "Effects of Postulated Event Devices on Normal Operation of Piping Systems in Nuclear Power Plants"(3) confirms these concerns. A rigid system requires great care in alignment to minimize expansion loads during heat-up and cool-down. Any binding results in stress intensification at the point of restraint, thus shortening fatigue life. The requirements to remove the test snubbers accentuates the problem The Licensing Event Reports confirm that misalignment and hence severe loads and overstress have occurred, since snubbers have pulled out of walls. An optimum piping system design must consider normal operational loads, accident loads, inservice inspection and As Low As Reasonably Achievable implications.

As a result of the above concerns, activities are underway to define a new approach which could greatly simplify design practices for seismic supports and thus significantly reduce commodity and manpower requirements while contributing to a safer system. The Pressure Vessel Research Committee (PVRC) has set up a committee to plan action on the general subject of "Design of Mechanical Systems for Dynamic Loadings, Procedures and Allowable Limits". Any results would need to be factored into the ASME Boiler and Pressure Vessel Code, regulatory guides and the standard review plans.

It is a sad commentary on our present regulatory system that plants currently under construction are spending millions of dollars to comply with requirements that are often felt to be overly conservative with regard to their treatment of a low probability event and potentially hazardous due to their impacts on normal operations. Examples of conservatism are present piping damping factors related to seismic design and the practice of using the double-ended pipe break design basis for determining structural loads. The projected time scale to remedy this situation is much greater than the time scale for introduction of new requirements.

"Many codes and standards were contrived and written by well-qualified, well-meaning individuals projecting ideal situations. They never had any idea that in this day and age of rigid quality assurance and quality control, the codes and standards would be enforced to the letter."(4) Using the non-exact solutions presented in the codes as input to complex analyses and then requiring the output to be considered an exact requirement is illogical. The complexity of engineering design to accommodate seismic events involves complex and interactive computer analysis involving various disciplines (i.e., earthquake ground motion, soils interaction, structural finite element and dynamic analysis in conjunction with elastic mechanical systems). The complexities of the resulting analyses are geared to 6-digit solutions. Experienced judgement is not allowed to affect the setting of clearances or tolerances generated by these analyses. Lacking such judgement when tolerances are set, judgement cannot be admitted when precision is legally required by QA inspectors and field installers. The attempt to accommodate a variety of possible tolerances could require a virtual infinity of analysis. Experience and judgement does not easily apply to low probability events. Conversely, where experience and judgement apply to higher probability events, it is ironic to note that the restraints governing the low probability events have taken precedence. Experienced field personnel have all but given up trying to change "impossible" requirements. It is as though the mind-set is commonplace that any method of doing something which is costlier, more difficult or more time-consuming is safer. These factors apply to concrete, electrical and other mechanical systems. We expect that subsequent case studies will provide a similar focus.

The concensus opinion in initiation of this case study was that regulatory requirements could be charged with the responsibility for increased costs, schedules and deteriorating productivity. In the conduct of this piping case study, it became clear that all parties bear responsibility in some degree for the current state of the industry. Designers have committed to approaches which are virtually impossible to achieve in the field. Architect engineers and associated construction forces are not accustomed to constructing or designing to machine type tolerances and in most cases are attempting field fit-up to a measured precision without adjustable shim provisions.

Management, engineers and construction personnel no longer question difficult criteria. If it is close to impossible to perform the activity is assumed to be in the interest of safety and impossible to change. Robert Eshbach of Gilbert Associates described the current atmosphere under which nuclear power plants are built:

> "The construction of nuclear power plants, and to a lesser extent the construction of other facilities, is controlled by a value system comprised, at least in part, by these tenets:
>
> 1. Avoid liability.
> 2. Accept only that responsibility which has been expressly delegated.
> 3. Perform work in accordance with detailed procedures approved by others so that if something goes wrong the blame can be spread.
> 4. Except in dire emergencies permit no exercise of judgement, common sense or any other unprogrammable human quality.
> 5. Establish as the highest project priority the faithful maintenance of records.
> 6. Do not offer advice or opinions to interfacing organizations. It can be used against you.
> 7. Take no action which could possibly endanger you as an individual or your career."

The past decade has witnessed the trend for industry to build staffs to mirror those of the federal systems (i.e., environmental, safety and analytical R&D staffs). The NRC and industry analytical staffs often interact to agree on an approach which defines impractical methods, tolerances, etc., enforcement and QA personnel are mandated to insist that the defined design be met exactly. It appears that neither the management of industry nor regulatory is currently structured to achieve a more practical approach to setting goals which can be realized. It is believed that the case study presented in this report provides insight on the legalistic approach to zero risk contributing to the currently unacceptable cost and schedule performance. The case offered here is that industry and government have collaborated to unwittingly specify impractical and unrealistic design goals which when attempted contribute to prohibitive costs and a potential decline in safety.

Mr. CHIPMAN. This particular example provides some insight into how problems of this kind can be minimized in the future through cooperation of safety and operational needs. In this case activities are underway within the industry and the NRC to define a new approach which could greatly simplify design practices for seismic supports and thus significantly reduce structural and manpower requirements. These all contribute to a safer system.

In conclusion, I believe better coordination of all safety research programs is required and significant progress is being made through the DOE working groups. This will result in improving the identification and prioritization of research and development requirements and improving the applicability and utilization of results toward assuring a continuing high level of safety for LWR's.

Public Law 96-567 provides a proper base for improving this coordination and the Department will, through this act, continue to work closely with the Commission and the industry to achieve the desired results. The results of our efforts will provide an important tool for the Congress to evaluate the various Government research and development programs in this country.

Thank you, Madam Chairwoman.

[The prepared statement of Mr. Chipman follows:]

Testimony of
Gordon L. Chipman, Jr.
Deputy Assistant Secretary for Nuclear Reactor Programs
U.S. Department of Energy
Subcommittee on Energy Research and Production
Committee on Science and Technology
U.S. House of Representatives
May 19, 1982

Madam Chairman, Members of the Committee, it is a unique privilege to be asked to testify before you today. I appreciate the opportunity to comment on the Nation's nuclear reactor safety research and development program.

My comments are derived mainly from the Department's interaction with NRC and the nuclear industry including National Laboratories, Department contractors, and universities in fulfilling the objectives of Public Law 96-567, the "Nuclear Safety Research, Development, and Demonstration Act of 1980," authored by this Subcommittee. It is this Act and the Department's efforts to respond to it that form the basis for my comments.

In establishing this Act, a major finding of the Congress was that, "continued development of nuclear power is dependent upon maintaining an extremely high level of safety in the operation of nuclear powerplants and on public recognition that these facilities do not constitute a significant threat to human health and safety." Accordingly, the Act provides for an accelerated and coordinated program of light water reactor (LWR) safety research development, and demonstration to be carried out by the Department.

The Act takes the broad view that safety improvements are to be considered in light of their costs and benefits and that complexity should be minimized in the process.

Because of the wide range of groups concerned about nuclear safety, the Congress at the urging of this subcommittee directed that the Department develop a coordinated safety R&D program, both with domestic and international groups. As you will see, we have diligently pursued these guidelines.

The Department believes that the objectives of Public Law 96-567 cannot be met by focusing separately on improving each of the commonly identified technology areas. Therefore, a comprehensive definition of program requirements is being developed to determine what R&D is appropriate in each technology area and what the relationships are between these individual technology areas and improving the overall safety of LWRs.

The Department has convened 10 working groups of representatives from the DOE, NRC, Institute for Nuclear Power Operations (INPO), Electric Power Research Institute (EPRI), the utilities' IDCOR program, utilities' owners groups, individual utilities, DOE contractors, the AIF, the EEI, the ANS, national laboratories and universities to assist the Department in developing the program requirements.

The overall objective of the DOE/NRC/Industry working groups is to (a) develop a comprehensive identification of what the LWR safety issues are and what efforts are required to resolve these issues, (b) review work that is underway to resolve these issues, (c) define programs that would resolve any remaining items of importance, and (d) assist the Department in developing a plan for prioritizing and implementing the programs.

In implementing these program activities, we plan to coordinate with the NRC and to the extent necessary, enter into new memoranda of understanding or revise existing memoranda for the purpose of eliminating unnecessary duplication and avoiding programmatic conflict.

The Department will also coordinate these activities with such other Government agencies, foreign governments and industry, as appropriate, to utilize their expertise, to minimize duplication of efforts, and to ensure that information useful for improved concepts applicable to nuclear powerplant safety can be applied in a timely manner. The Department will enter into agreements or memoranda of understanding, as required, with these organizations.

A major output from the working groups will be identification of safety R&D areas warranting attention in order of relative priority.

Approximately 150 experts from 40 organizations are contributing members and the NRC has been working closely and most cooperatively in these efforts. The process of completing the working group tasks is a difficult and time consuming one, primarily because of the wide range of views involved and the newness of this concept of coordinated R&D. It is important that this process be completed in a thorough, disciplined manner in order to assure that the end result will be both technically sound and institutionally accepted.

These efforts should provide a most useful tool for the Congress, the Department, NRC, and industry in identifying and satisfying LWR safety requirements.

Based on these efforts and review of NRC program plans, I believe that significant improvements can be made in existing safety R&D programs in the areas of:

- Program integration;
- Interaction with other programs and ultimate users of the results;
- Identification of how R&D results will be used to resolve issues;
- Identification and prioritization of R&D needs; and
- Applicability and increased utilization of R&D results.

Our efforts in responding to Public Law 96-567 have led us to identify a very important goal of effecting better coordination of regulatory and operational requirements. Historically the NRC has focused its research efforts on demonstrating or confirming the conservatism of its regulations. Its role as an independent agency has naturally led to a degree of isolation from outside views challenging this narrow focus and consequently from many of the operational and potential safety implications of this approach. As a result of the TMI accident, while the general level of safety of nuclear power plants was demonstrated, improvements have been achieved in broadening the view of what is of safety significance. It appears, however, that further improvements in the area of safety R&D, including coordination among all interested parties and simplification of requirements, are appropriate.

I would like to expand briefly on what I mean by this.

The purpose of safety research programs should be to provide the information necessary to assure an acceptable level of risk to the public from operation of nuclear powerplants. In this regard there are two important ways in which the risk of plant operations may be improved.

(1) Insuring that the plant design and operation address the maximum range of potential accidents, and

(2) Improving the reliability of plant operations.

A safety R&D program which balances these considerations would be most effective in improving reactor safety. We believe the historical lack of coordination of R&D programs by the developmental, design, and regulatory organizations has led to an over emphasis on the first element and insufficient attention to the second.

The Department is seeking through its obligations under P.L. 96-567 to improve the effectiveness of safety R&D. We believe the efforts of all parties in the Working Groups established by the Department have overcome many past barriers to cooperation and provide the means to develop a truly balanced safety R&D program.

One manifestation of the historical approach to safety research is the major emphasis placed on the development of refined analytical techniques which can calculate effects at very low probabilities. This capability has little or no relationship to actual construction or operational needs and practices.

Consequently, when modifications to design and operation have been made to provide apparent safety improvement as a result of this refined analysis capability, the reliability of plant operations can be impacted with potential safety implications. In addition, the combination of safety through conservatism and focus on low probability events leads to requirements based on poorly understood phenomena, resulting in cost increases and potential safety concerns.

The Department has investigated in considerable detail one such impact which we believe to be typical of the kind of problem which can result. The example has to do with pipe restraint analysis techniques and design criteria which have been developed to provide accommodation for low probability earthquakes and other low probability piping loads. This analytical emphasis on low probability concerns has resulted in severe detrimental impacts to operating nuclear powerplants.

During the course of this investigation, it became clear that the practice of restraining piping to minimize movement during earthquakes results in a system which is not necessarily compatible with the requirements for thermal expansion and contraction during normal operation. There is growing concern that the current design practice of severely stiffening piping systems is contributing to the potential for overstress due to thermal expansion. A rigid system requires great care in alignment to minimize expansion loads during heat-up and cool-down. Any binding results in stress intensification at the point of restraint, thus shortening fatigue life. Licensing Event Reports confirm that misalignment and hence severe loads and overstress have occurred. Pipe snubbers have pulled their anchor devices right out of walls from such loads. We have traded away one form of uncertainty (inability to accurately calculate stress in less constrained pipe) for a known problem of rigidity.

In addition, the large number of hangers, seismic supports and pipe whip restraints have resulted in crowding areas with structural members. Access to piping (and other systems) is restricted, making it more difficult for both men and equipment to be brought to the work site. Maintenance activities are also severely constrained.

An optimum piping system design must consider normal operational loads, inservice inspection and As Low As Reasonably Achievable (ALARA) occupational dose implications as well as accident loads.

I am providing a copy of the executive summary of our report on this matter which provides additional information. The complete report is also available for anyone interested.

I believe that this example also provides insight into how problems of this kind can be minimized in the future through coordination. As a result of the above concerns, activities are underway within the industry and NRC to define a new approach which could greatly simplify design practices for seismic supports and thus significantly reduce structural and manpower requirements while contributing to a safer system. The Pressure Vessel Research Committee (PVRC), made up of industry and NRC members, has set up a group to plan action on the general subject of "Design of Mechanical Systems for Dynamic Loadings, Procedures and Allowable Limits."

In other words the problem is being addressed by a better coordination between NRC and industry of regulatory, safety, and operational needs. We are hopeful that this effort will result in more viable requirements.

In conclusion, I believe better coordination of all safety research programs is required and significant progress is being made through the DOE working groups. This will result in improving the identification and prioritization of R&D requirements and improving the applicability and utilization of R&D results toward assuring a continuing high level of safety for LWRs. Public Law 96-567 provides a proper base for improving this coordination, and the Department will, through this Act, continue to work closely with NRC and the industry to achieve the desired results. The results of our efforts will provide an important tool for the Congress to evaluate the various government R&D programs.

Mrs. BOUQUARD. Thank you very much, Mr. Chipman.

Going back to Public Law 96-567, which does direct the Department to establish a comprehensive program on LWR safety research development and demonstration, in coordinating our activities which we have been discussing yesterday and today, how do you really view the role of the Department of Energy in carrying out this mandate?

Mr. CHIPMAN. I believe the most important role that the Department has is to bring together the various parties that have interest and responsibility in the area of reactor safety—that includes the utilities, reactor vendors, the research organizations, the regulatory body of the Nuclear Regulatory Commission, and the Department of Energy itself—to make sure that all of the important factors that need to be taken into account to make logical, efficient, cost-effective safety decisions are in fact brought together. That is the process we are in right now.

We can play a larger or lesser part in the amount of research that we actually conduct within that role.

Mrs. BOUQUARD. Do you look upon the role of DOE as supplying the data that will be assessed by NRC in this area?

Mr. CHIPMAN. We will, in response to the act, provide the Congress and others, including the Nuclear Regulatory Commission, the findings of the working groups and the Department in putting this information together. I believe it will be useful for all parties to assist them in deciding where the efforts in safety research should be focused.

Mrs. BOUQUARD. Do you feel that some of your safety related activities by your 10 working groups will be covered by current programs at NRC? What interaction do you envision in this area between NRC and DOE?

Mr. **Chipman**. Many of the areas that we are looking at are covered in one way or another by the rather extensive NRC safety research program. I believe, however, that our final results may shed different light on the priorities that should be applied to these areas as well as the specific focus within a given research area. We may find that a different emphasis should be placed.

Mrs. **Bouquard**. I would like for you to expand a little bit on the statement that you made.

You mentioned lack of complexity as an objective in safety systems. Could you give me some examples to redefine this for me?

Mr. **Chipman**. I think the one that I gave in the testimony is one case of that. I talked about how we end up with a piping system that is covered with restraints and seismic devices to limit its movement. This, in one fashion, is a complexity that has been added to the system. It results in a lot of problems.

There are others though that are perhaps a little more dramatic. As we found out in the accident at Three Mile Island, the release of the free form of iodine when there is severe core damage does not seem to occur. However, in years past, requirements were placed on power plants to install systems to remove free iodine from the atmosphere within the containment. These systems inject liquid into the containment atmosphere following an accident to remove the iodine.

At Three Mile Island 1, the plant that did not have the accident, the system happened to contain sodium thiosulphate and some time ago this system got mixed up and some of this chemical got injected into the primary system resulting in the tube failures in the steam generators. Therefore we have a case for the safety system—the spray system—a complex, costly system—that was installed with the use of chemicals to cover an accident situation which, if even possible, is extremely unlikely to occur in real life—has had negative safety implications. It has actually caused severe damage to the plant.

Mrs. **Bouquard**. Going back to the term "low probability" I would like you to quantify for us the range of probabilities that you are implying here. What do you really consider "low probability"?

Mr. **Chipman**. I am uncomfortable with trying to specify a range of probability that is appropriate to focus on. I think perhaps a more fruitful way to look at the research effort should be that there are a number of basic functions that we want to maintain in any powerplant.

First of all we want the plant to operate safely. Second of all, we want to protect the core. Third of all, we want to make sure that the containment integrity is maintained if the core should be damaged. Last, we are interested in making sure that we can take appropriate action to safeguard and evacuate, if necessary, people in the area of a plant if containment integrity cannot be maintained.

If we take each of the basic ideas of what it is we are trying to accomplish and for each one look at the more likely events that might jeopardize it, I think we might avoid many of the extremely complex, unlikely and perhaps impossible scenarios that have developed.

Let me give you one example of how I think we might approach this. If we simply say that we want to make sure that if the core is damaged—we take that as a given—if the core is damaged we want to maintain containment integrity, the idea then is to look at maintaining integrity in the more likely ways that might be jeopardized rather than finding such extremely low probability events that they cannot be practically considered.

What you have really done then is to say, "I have an unlikely event to begin with, that is, core damage which has very low probability, and instead of examining the most likely threats to a breach of containment, I have looked at perhaps one in a hundred events that might jeopardize containment integrity."

In this way I think we might avoid many unrealistic scenarios.

Mrs. **BOUQUARD**. Unfortunately we have bells. We will stay in recess until the Chair can return.

[Whereupon the subcommittee stood in recess at 2:16 p.m. to 2:33 p.m.]

Mrs. **BOUQUARD**. We will resume our hearings.

One final question, Mr. Chipman, would you describe for us the relationship between NRC and DOE research efforts on breeder reactors? What I am really concerned about is do you feel there is unnecessary duplication of efforts in this program?

Mr. **CHIPMAN**. As a piece of background on that, there is a vast difference in the research efforts of the two departments in breeder reactors as compared to light water reactors.

The major effort in the breeder area is with the Department of Energy and the Nuclear Regulatory Commission has a relatively small effort.

I do not believe there is really any significant duplication here that is not reasonable. However, we have run into a problem that is related to this. That is, whenever you have a new project that comes to the front, people see an opportunity I think to get their finger in the pie and make money on it. There have been a lot of interests outside the Department and outside the Nuclear Regulatory Commission, looking for problems to solve related to the breeder reactor.

I think this has been causing us some difficulties. We are going to overcome it but it is a problem.

Mrs. **BOUQUARD**. Our testimony yesterday reached the conclusion that much of the work that has been done by DOE and by industry certainly is sufficient for providing NRC with the background material that they need in addition to the information at NRC—the data they have already accumulated.

Mr. **CHIPMAN**. I would agree that we believe that is true. The only qualifier is that the NRC program is not particularly large, as it is, in breeder areas. The potential negative outfall, however, is that their programs will be focused on extremely conservative criteria which will, in the end, slow the program and increase costs.

Mrs. **BOUQUARD**. What recommendations do you have in this area?

Mr. **CHIPMAN**. I would suggest that it be looked at particularly by the Congress to see to it that there is not unnecessary duplication.

Mrs. **BOUQUARD**. Thank you very much for your testimony. We certainly appreciate having you with us today.

Mr. **CHIPMAN**. Thank you, Madam Chairwoman.

Mrs. **BOUQUARD**. Thank you very much.

Our next witness is Dr. John Taylor, director of the nuclear power division, Electric Power Research Institute.

We are happy to have you with us again today, Dr. Taylor. It is a pleasure to have you appear before our committee and we do welcome you.

The full text of your prepared statement will be included in the record and you may summarize if you wish.

STATEMENT OF DR. JOHN TAYLOR, DIRECTOR, NUCLEAR POWER DIVISION, ELECTRIC POWER RESEARCH INSTITUTE

Mr. TAYLOR. Thank you, Madam Chairwoman.

I am John Taylor, director of the nuclear power division, Electric Power Research Institute. I will summarize my comments since I have submittted my statement for the record.

Mrs. BOUQUARD. Proceed as you wish. Your testimony has been made a part of the record.

Dr. TAYLOR. Thank you.

NRC has been most cooperative in supplying us information on their research plans and associated budgets. Our understanding is that the draft documents we have received are in the process of being revised.

The breadth of the NRC plan is impressive and, if properly focused, the work will be beneficial. The NRC staff has worked hard accumulating information and planning their research. It is our hope that our discussion with you today will be in the form of a dialog with the understanding that we will emphasize the improvements that can be made rather than the planning segments that have been well done. Although we are concentrating on needed improvements it should be understood that in our continuing dialogue with NRC it is clear that NRC recognizes the need for these improvements and has already taken steps toward such improvement as we compare the most recent plans with those of a year ago.

In general, NRC's program needs to be prioritized and then formulated around specfic deliverables relating to resolution of safety issues or confirmation of safety. The NRC program would accomplish more with greater speed if the resources were focused in this manner. This recommendation reinforces that made by the ACRS in their report to Congress in NUREG-0864.

The philosophic basis for the program is of concern to us. It is based on the implicit assumption that if the codes and analytical methods which make complex mechanistic predictions of low probability events are upgraded by numerical, analytical and "input" improvements the issues will be resolved.

While this approach is commendable and necessary, it is not sufficient. Carefully designed demonstration tests of large or modest scale are needed to provide credible benchmarks for "conclusions" predicted by the computer codes. Without a few such well thought-out experiments, the research and development program may well be limited to providing input to complex and uncertain predictions that will remain uncertain.

For low probability events associated with degraded core cooling, the major need is for experimentally validated computer codes which will adequately predict: first, the release magnitudes of risk-significant radionuclides to the atmosphere; second, the probable location of radionuclides within the primary coolant system and containment following a severe accident; and third, the limits within which severely degraded cores can be cooled.

The first provides a more realistic basis for emergency response planning and the necessary confidence that a plant meets, or does not meet, NRC's safety goals. The second provides a more accurate basis for equipment qualification, the proper selection of filters and design of residual heat removal systems, etc. The third provides improved guidance for the management of severe accidents and the preparation of emergency procedures, etc.

Addressing the above points does not require detailed modeling of the behavior of melting fuel assemblies. We recommend that this type of work be de-emphasized and NRC focus on resolvable technical issues which will provide sound regulatory standards.

Exhibit 1 is given as an example of our judgment of resolvable technical issues associated with the magnitude and modes of fission product releases in the event of a severe core accident. We derived this list through evaluation to determine the relative importance and impact of uncertainties which exist in the picture at this time.

We then developed exhibit 2 which displays EPRI's research program and schedule in the degraded core and source term areas which would direct itself at resolution of those technical issues.

As these exhibits suggest, the majority of these issues could be resolved by fiscal year 1984 and it is our judgment that all could be resolved through fiscal year 1985.

A last general comment is that the draft plans principally address mitigation research for severe core damage. The NRC should aggressively pursue improvements in technology for prevention of severe accidents. The draft program plans are disappointing in their apparent deemphasis of this very important technical and regulatory area.

I would like now to summarize our comments on the decision units in the NRC research and development plan.

In the "LOCA and transient" decision unit, NRC should increase its efforts at integration and coordination of computer code development and experimental programs, giving high priority to decay heat removal studies. NRC has concentrated on producing large, complex computer codes for reactor safety analysis. The time is right for consolidation of this effort and its integration with industry programs.

Increased emphasis on the actual, realistic evolution of accident sequences is required. The need for an independent assessment capability for NRC is laudable but NRC's computer codes are mostly suitable for only very restricted transients of extreme nature.

Assurance of the validity of NRC's assessments can be achieved by benchmarking and other formal computer code validation processes.

The LOFT decision unit is specific to a particular facility. We have been disappointed in the past that the results of LOFT tests have not been clearly and forcefully introduced into the licensing arena. The LOFT results have been used by the NRC staff to confirm the conservatism of their overall safety decisions. This and other NRC research has not been used to reduce unnecessary conservatisms in the regulations where it has been demonstrated that a change in the regulations will not affect the safety of nuclear plants. The impact of some regulations is such as to inhibit cost effective electricity generation while not contributing to safety.

There are a number of technical areas where LOFT might make a contribution. These include the study of degraded core cooling, testing the actual safety limits of current fuel design as opposed to the fuel design limits defined in the licensing process, and the transport and retention of fission products through the nuclear reactor system.

The NRC program on accident evaluation and mitigation does not address prioritization of researchable items; it attempts to cover many aspects of postulated and extremely low probability degraded core accidents, not all of which are important to public safety.

It is recommended that event tree analysis from probabilistic risk assessment studies be used as input to the NRC plan to focus research efforts on those aspects of degraded core accident scenarios that lead to the greatest risk to the health and safety of the public.

It is important to focus on these crucial items so that the Accident Evaluation and Mitigation program can reach conclusions and permit decision-making in the 1983–85 period. To do this, NRC should make continued use of foreign and EPRI funded work integrated into the NRC plan.

We strongly recommend NRC's participation with EPRI in an international consortium which will conduct large scale experiments of the aerosol behavior in the primary coolant system at Marviken. Thus, the NRC can leverage moneys spent to conduct research on the behavior of aerosols in the primary system following a severe core accident.

In summary, we are concerned with the diffuse nature of this program area and would agree in principle with ACRS that "The proposed Severe Accident Research Plan is composed primarily of expensive, long-term experiments whose results we believe are not needed to resolve the major issues related to severe accidents."

In the systems and reliability analysis unit, strengthening the risk/benefit methodology and risk analysis basis is not clearly established as an objective. A plethora of probabilistic risk assessment analyses are currently planned in order to gain "technical insights." Many of these PRA's have already been done by industry, and additional repetitive PRA studies are not likely to gain markedly new insights.

What is recommended are "normalized" analyses that compare competing methodologies of different groups and organizations on a basis that allows a direct comparison that is technically valid.

In the siting and environment decision unit, we fully endorse NRC's objectives for seismic research, that is; the developing of a more realistic earthquake ground motion specification, the assessing of the sensitivity of seismic risk to uncertainties in various elements of the seismic design chain, and the validation of techniques to estimate conservatisms in seismic safety licensing requirements.

We support NRC research recommendations for changes in licensing criteria to more realistically account for structural and system damping. It is important to quantify information on seismic risk and the rank of the components contributing to the seismic risk so that safety issues associated with earthquake considerations do not result in overdesign and can be effectively resolved.

In the reactor and facility engineering decision unit, we propose that, in the area of equipment qualification, cooperative efforts be initiated using probabilistic risk assessment to quantify the cost and benefit of increasingly restrictive regulations, many of which are based on the findings of limited testing conducted at a national laboratory under NRC sponsorship.

The expenditure of industry resources on higher priority issues may provide greater improvements to safety. For example, it is not clear that automatic extension of existing electrical equipment requirements to mechanical equipment, whose qualification is backed by a century of technological improvement under the ASME, would lead to an improvement in safety.

In the facility operations and safeguards unit, emergency preparedness is an important area which can have direct public impact. NRC has not yet accounted for the improved, and we believe more realistic, assessment of fission product source term in the analysis of severe accident scenarios for the emergency planning rule. It is important to determine the source term of fission products as accurately as possible and to implement regulations and decisions on this basis to protect the safety of neighbors to nuclear power plants.

Too high a source term determination will result in taking emergency action dangerous to the public when not needed. Too low a source term will result in not taking emergency action when needed. Taking action when not needed will expose the neighbors of a nuclear plant to evaculation harards and unnecessary mental anguish, resulting in nonnuclear risks which might be greater than nuclear risks.

In summary, the NRC research and development plan is impressive and crucial to maintaining the excellent safety record of the nuclear power industry. However, it should be redefined to first identify priorities, better define milestones and tighten up schedules. Second, it should relate more clearly the technical programs to the safety issues and the decision making processes and schedules to resolve these issues. Third, it should establish goals of software validation through selective, key experimental programs. Fourth, it should include goals for the reduction of undue conservatism to achieve a more balanced approach to safety, and fifth, it should increase the emphasis on preventing degraded core cooling as contrasted to mitigating degraded core cooling accidents.

This last point is of crucial importance to the utility industry and underscores the common goal which it shares with NRC. Maintaining core cooling assures NRC success in meeting its goal of protecting the health and safety of the public. It also assures each utility's success in protecting the health and safety of the public and the utility's workers and the utility's major investment in the nuclear facilities it operates.

Thank you very much.

[The prepared statement of Dr. Taylor follows:]

PREPARED STATEMENT OF DR. JOHN J. TAYLOR, DIRECTOR, NUCLEAR POWER DIVISION, ELECTRIC POWER RESEARCH INSTITUTE

I am John J. Taylor, Director of Nuclear Power, the Electric Power Research Institute. I appreciate the opportunity to comment on the Nuclear Regulatory Commission's (NRC) R&D planning.

NRC has been most cooperative in supplying us information on their research plans and associated budgets, in particular DRAFT NUREG-0900 "Nuclear Plant Severe Accident Research Plan," April 2, 1982, "Long Range Research Plan FY 1984-1988," Draft NUREG-0784, March 15, 1982, and other personal communications with the EPRI staff. Our understanding is that the draft documents we have received are in the process of being revised.

The breadth of the NRC plan is impressive and, if property focused, the work will be beneficial. The NRC staff has worked hard accumulating information and planning their research. It is our hope that our discussion with you today will be in the form of a dialogue with the understanding that we will emphasize the improvements that can be made rather than the planning segments which have been well done. Although we are concentrating on needed improvements, it should be understood that in our continuing dialogue with NRC it is clear that NRC recognizes the need for these improvements and has already taken steps toward such improvement as we compare the most recent plans with those of a year ago.

General Comments on Program

In general, NRC's program needs to be prioritized and then formulated around specific deliverables relating to resolution of safety issues or confirmation of safety. The NRC program would accomplish more with greater speed if the resources were focused in this manner. This recommendation reinforces that made by the ACRS in their report to Congress (NUREG-0864).

The philosophic basis for the program is of concern. It is based on the implicit assumption that if the codes and analytical methods which make complex mechanistic predictions of low probability events are upgraded by numerical, analytical and "input" improvements the issues will be resolved. While this approach is commendable and necessary, it is not sufficient. Carefully designed demonstration tests of large or modest scale are needed to provide credible benchmarks for "conclusions" predicted by the computer codes. Without a few such well thought-out experiments, the R&D program may well be limited to providing "input" to complex and uncertain predictions that will remain uncertain.

For low probability events, the major need is for experimentally validated computer codes which will adequately predict: (1) the release magnitudes of risk-significant radionuclides to the atmosphere; (2) the probable location of radionuclides within the primary coolant system and containment following a severe accident; and (3) the limits within which severely degraded cores can be cooled. The first provides a more realistic basis for emergency response planning and the necessary confidence that a plant meets (or does not meet) NRC's safety goals. The second provides a more accurate basis for equipment qualification, the proper selection of filters and design of residual heat removal systems, etc. The third provides improved guidance for the management of severe accidents and the preparation of emergency procedures, etc.

Addressing the above points does not require detailed modeling of the behavior of melting fuel assemblies. Modeling in detail the melting of cores and an experimental program to support that modeling is technically very difficult and the current NRC program is not finely focused enough to be cost-effective for the results achieved. We recommend that this type of work be de-emphasized and that NRC focus on resolvable technical issues which will provide sound regulatory standards. Exhibit 1 is given as an example of our judgement of resolvable technical issues associated with the magnitude and modes of fission product releases in the event of a severe core damage accident. Exhibit 2 displays EPRI's research in the degraded core and source term areas. As these exhibits suggest, a majority of these issues could be resolved by FY 1984; and it is our judgement that all could be resolved through FY 1985.

This schedule is recommended for all degraded core related research; that is, significant interim accomplishments in two years and wrap-up by the end of FY1985.

A last general comment is that the draft plans principally address mitigation research for severe core damage. The NRC should aggressively pursue improvements in technology for prevention of severe accidents. The draft program plans are disappointing in their apparent de-emphasis of this very important technical and regulatory area.

I would like now to summarize our comments on the Decision Units in the NRC R&D Plan.

LOCA and Transient

In the "LOCA and Transient" Decision Unit, NRC should increase its efforts at integration and coordination of computer code development and experimental programs, giving high priority to decay heat removal studies. A more restricted involvement in the Upper Plenum Test Facility (UPTF) in Germany beyond 1984 should be considered because of program delay, and increases in projected costs without a corresponding increase in useful information. If further data are felt necessary by NRC, consideration should be given to performing smaller-scale UPTF tests in the United States. EPRI, NRC, and the reactor manufacturers are co-sponsoring three important programs funded in the LOCA and Transient Decision Unit: (1) the Boiling Water Reactor (BWR) Full Integral Simulation Test (FIST) program which will lead to an improved understanding of the reactor transient response and the effectiveness of emergency core cooling in the event of small pipe break loss of coolant accidents in BWRs; (2) a similar program (FLECHT-SEASET) for the pressurized water reactor (PWR) systems; and (3) a joint large-scale testing program to evaluate thermal mixing and thermal resistance during PWR high pressure injection of emergency cooling water into the reactor primary system. This is of direct importance to the assessment of reactor pressure vessel integrity under the combined burden of thermal shock and cold over-pressurization.

NRC has concentrated on producing large complex computer codes for reactor safety analysis. The time is right for the consolidation of this effort and its integration with industry programs. Increased emphasis on the actual realistic evolution of accident sequences is required. The need for an "independent assessment" capability for NRC is laudable, but NRC's computer codes are mostly suitable only for very restricted transients of extreme nature. Assurance of the validity of NRC's assessments can be achieved by "benchmarking" and other formal computer code validation processes.

LOFT

The LOFT Decision Unit is specific to a particular facility. We have been disappointed in the past that the results of LOFT tests have not been clearly and forcefully introduced into the licensing arena. The LOFT results have been used by the NRC staff to confirm the conservatism of their overall safety decisions. This and other NRC research has not been used to reduce unnecessary conservatisms in the regulations where it has been demonstrated that a change in the regulations will not affect the safety of nuclear plants. The impact of some regulations is such as to inhibit cost-effective electricity generation while not contributing to safety.

For example, LOFT has clearly demonstrated physical phenomena (rewetting in blowdown and confirmation of quick refill) not considered in LOCA licensing regulations. If these results were utilized in licensing, plant operators could improve plant capacity factor and operate more efficiently with fewer operating restrictions. Research from other decision units which is being ignored include decay heat and zircaloy oxidation results which also suggest modifications of licensing basis leading to less restrictive peak fuel power ratings.

Because of this background, the industry is not enthusiastic about a continuing LOFT program. If, on the other hand, LOFT had an impact on the licensing of nuclear plants, EPRI would encourage means by which LOFT can be kept available as a research facility. Recent discussions on the formation of an international consortium are relevant to this. There are a number of technical areas where LOFT might make a contribution. These include: study of degraded core cooling; testing the actual safety limits of current fuel design as opposed to the Fuel Design Limits defined in the licensing process; and the transport and retention of fission products throughout the nuclear reactor system.

Accident Evaluation and Mitigation

The NRC program on Accident Evaluation and Mitigation does not address prioritization of researchable items. It attempts to cover many aspects of postulated (extremely low probability) degraded core accidents, not all of which are important to public safety. It is recommended that event tree analysis from Probabilistic Risk Assessment studies be used as input to the NRC plan to focus research efforts on those aspects of degraded core accident scenarios that lead to the greatest risk to the health and safety of the public.

It is important to focus on these crucial items so that the Accident Evaluation and Mitigation program can reach conclusions and permit decision-making in the 1983-1985 period. To do this NRC should make continued use of foreign and EPRI funded work integrated into the NRC plan. We were disappointed to find apparent cases of old continuing projects remaining in the research plan with some redirection of the effort but without apparent objective consideration of their contribution to a priority technical question.

We observe that about 50% of the Accident Evaluation and Mitigation Funds are earmarked for the behavior of degraded fuel. Research on fuel damage of and by itself is not important without attention to fission product release, transport, and behavior which has impact on the public health and safety. The tests for the National Research Universal (NRU) reactor at Chalk River, Canada, the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory (INEL) at Idaho Falls, Idaho, and the Annular Core Research Reactor (ACRR) at Sandia Laboratorie in Albuquerque, New Mexico, seem more than is needed. It is recommended that NRC review these fuel tests and re-program to achieve unambiguous, experimental benchmarks.

The computer codes that predict the events during severe core damage accidents need strengthening at the fundamental level. It is not clear from the plan that the scientific basis of these mechanistic consequence codes will be adequately examined. Detailed technical reviews have recently been sponsored by EPRI and NRC. An aggressive follow-up of these studies to improve these computer codes is strongly recommended.

An underlying problem in the definition of research programs for "Accident Evaluation and Mitigation" is that the experimental conditions to be investigated are defined using a computer code (MARCH) that is not technically realistic. Prediction of unduly severe and extreme conditions leads to experiments that are needlessly expensive, time consuming and not necessarily relevant to reactor safety. In the development of the SCDAP code (on Severe Core damage fuel behavior), NRC appears to be going into more detail (and expense) than is justifiable to understand the progression of a severe core accident. From the thrust of NRC's modeling program, the link (either direct or indirect) between reactor safety and computer code improvements is not clear. The plan should concentrate on the development of experimentally validated computer codes that are needed to predict the release magnitude of risk-significant radionuclides to the atmosphere, the location of radionuclides within the primary coolant system and containment following a severe accident, and the limits within which a severely degraded core can be cooled. These items would contribute to a more realistic basis for emergency planning, an improved basis for equipment to withstand accident conditions and function during accidents, and an improved guidance for management of severe accidents and preparation of emergency procedures.

We strongly recommend NRC's participation with EPRI in an international consortium which will conduct large scale experiments of the aerosol behavior in the primary coolant system at Marviken. Thus, the NRC can leverage monies spent to conduct research on the behavior of aerosols in the primary system following a severe core accident.

In summary, we are concerned with the diffuse nature of this program area and would agree in principle with ACRS (NUREG-0864) that "The proposed Severe Accident Research Plan is composed primarily of expensive, long term experiments whose results we believe are not needed to resolve the major issues related to severe accidents. The program in this Decision Unit should be restructured...This would allow the elimination of a substantial portion of the longer term experimental and code development work."

Finally, we strongly recommend that NRC produce a detailed plan for Human Factors similar to DRAFT NUREG 0900 (Severe Accident Research Plan) and address preventive measures in the current draft plan on Severe Accident Research.

System and Reliability Analysis

Strengthening the risk/benefit methodology and risk analysis basis is not clearly established as an objective. A plethora of Probabilistic Risk Assessment (PRA) analyses are currently planned in order to gain "technical insights." Many of these PRAs have already been done by industry, and additional repetitive PRA studies are not likely to gain markedly new insights. What is recommended are "normalized" analyses that compare competing methodologies of different groups and organizations on a basis that allows a direct comparison that is technically valid.

The containment failure mode studies proposed are potentially important, but the plan does not appear to give attention to potential scaling problems and has a rather relaxed schedule.

Siting and Environment

We fully endorse NRC's objectives for seismic research: that is, the developing of a more realistic earthquake ground motion specification, the assessing of the sensitivity of seismic risk to uncertainties in various elements of the seismic design chain, and the validation of techniques to estimate conservatisms in seismic safety licensing requirements. We support NRC research recommendations for changes in licensing criteria to more realistically account for structural and system damping. It is important to quantify information on seismic risk and the rank of the components contributing to the seismic risk so that safety issues associated with earthquake considerations do not result in overdesign and can be effectively resolved. EPRI has conducted simulated earthquake experiments and in-plant vibration tests as well as has developed computer codes in addressing seismic design issues. These research results and our on-going program will complement NRC's effort in improving our understanding of seismic design and seismic safety.

Reactor and Facility Engineering

For the Reactor and Facility Engineering Decision Unit, we have no direct comments on the program but have suggestions for increased NRC/EPRI technical cooperation.

We propose that in the area of equipment qualification cooperative efforts be initiated using probabilistic risk assessment to quantify the cost and benefit of increasingly restrictive regulations, many of which are based on the findings of limited testing conducted at a national laboratory under NRC sponsorship. The expenditure of industry resources on higher priority issues may provide greater improvements to safety. For example, it is not clear that automatic extension of existing electrical equipment requirements to mechanical equipment, whose qualification is backed by a century of technological improvement under the ASME, would lead to an improvement in safety. EPRI could provide input ensuring that NRC research conclusions lead to realistic requirements.

It is also recommended that EPRI review and have the opportunity for technical input into NRC criteria development for decommissioning of facilities. It is prudent to identify needs for technical support for site disposition decisions and actions well in advance of the need.

Facility Operations and Safeguards

Under the Facility Operations and Safeguards Decision Unit, we propose to upgrade an informal exchange to a systematic formal report exchange in the Human Engineering area between NRC(RES) and EPRI. EPRI's interest is not only in safety but also in the efficiency and effectiveness of control and diagnostic systems and devices.

Emergency preparedness is an important area which can have direct public impact. NRC has not yet accounted for the improved (realistic) assessment of fission product source term in the analysis of severe accident scenarios for the emergency planning rule. It is important to determine the source term of fission products as accurately as possible and to implement regulations and decisions on this basis to protect the safety of neighbors to nuclear power plants. Too low a source term will result in not taking emergency action when needed. Too high a source term determination will result in taking emergency action dangerous to the public when not needed. Taking action when not needed will expose the neighbors of a nuclear plant to evacuation hazards and unnecessary mental anguish, resulting in non-nuclear risks greater than nuclear risks.

Conclusion

In summary, the NRC R&D plan is impressive and crucial to maintaining the excellent safety record of the nuclear power industry. However, it should be re-defined to (a) identify priorities, better define milestones and tighten up schedules; (b) relate more clearly the technical programs to the safety issues and decision making processes and schedules to resolve these issues; (c) establish goals of software validation through selective, key experimental programs; (d) include goals for the reduction of undue conservatism to achieve a more balanced approach to safety; and (e) increase the emphasis on preventing degraded core cooling as contrasted to mitigating degraded core cooling accidents. This last point is of crucial importance to the utility industry and underscores the common goal which it shares with NRC. Maintaining core cooling assures NRC success in meeting its goal of protecting the health and safety of the public; it also assures each utility's success in protecting the health and safety of the public *and* the utility's workers *and* the utility's major investment in the nuclear facilities it operates.

RESOLVABLE TECHNICAL ISSUES

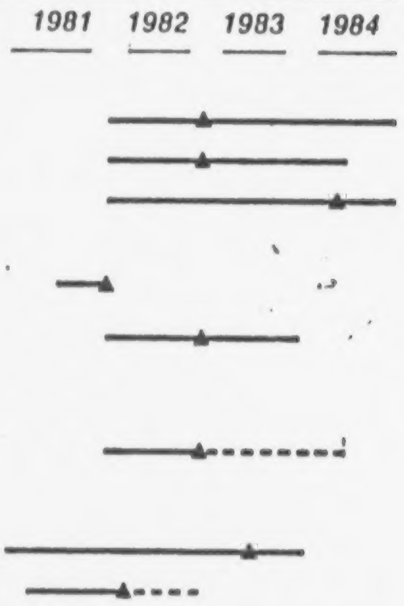
Fission Product Source Term Conservatisms	*Potential Reduction Factors*	*Technical Basis Within Two Years*
Quantity and timing of release of "volatile" fission product from fuel		
Chemical form of fission product	2–10	M
Lack of fission product retention in PCS (aerosol deposition, agglomeration and condensation)	2–10	M
No fission product trapping in steam-saturated quench tanks or suppression pools	10–100	L
Incomplete aerosol modeling for containment (aerosol agglomeration, steam condensation, deposition and settling)	2–10	H
Longer containment confinement period	2–10	M
No fission product deposition in containment leak passages		
Puff discharges at containment overpressure failure	2–5	L
No fission product retention by auxiliary buildings	2–5	L
	1.5–2	M

H = high probability; M = medium probability; L = low probability

Exhibit 1

CURRENT EPRI SOURCE TERM R&D PROGRAM
Focused Upon Risk Dominant Accident Sequences

Projects	*Objectives*
Fission product/aerosol retention in PCS	
• Primary source characterization	Establish likely fission product retention in PCS
• Thermal/hydraulic model	
• Large-scale aerosol behavior	
Water scrubbing	
• Model development	Establish likely fission product retention in water pools
• Medium and large-scale experiments	
Aerosols in containment	
• Validate codes against NRC experiments	Establish likely behavior of aerosols within containment
Duration of containment integrity	
• Debris bed experiments	Quantify steam "spike" and timing of containment integrity loss
• *MARCH* upgrade	



▲ First usable data

Exhibit 2

Mrs. **Bouquard**. Thank you very much, Dr. Taylor. We certainly appreciate your testimony. I want to read it further. It is certainly very fine.

I would like you to give some examples, if you would, to clarify your statement about NRC's emphasis on mitigation research.

Dr. **Taylor**. One area of substantial emphasis in mitigation research in the program is the development of a device which is popularly called a core catcher which would be put in place to prevent the spread of melted core materials in the event of a very severe accident involving a substantial core melt.

It is our judgment that the ability to identify the course of events in that improbable a condition is so limited that one could easily question the efficacy of such a mitigation device. There are many ways in which effort could be put forth to assure prevention of the accident or to assure a more limited extent of degraded core conditions in the event of a loss of cooling which would be substantially more effective and result in a greater safety position for the industry.

Mrs. **Bouquard**. Thank you very much.

In your statement on NRC computer codes, do you feel that they are most suitable for restricted transients of extreme nature—I believe that was your statement? Do you feel that they are not actually relevant in so many areas?

Dr. **Taylor**. If one looks at the course of development in computer codes, one finds that the industry has concentrated very heavily on the development of computer codes which describe the behavior of systems in their normal conditions and under moderate transients.

Much of the experimental validation of these codes therefore has been possible through the utilization of actual field and operating experience. The NRC program, which by its nature is looking at the low probability events, also has developed computer codes which attempt to describe the behavior of the system under unusual and extreme transients of low probability. These computer codes also need experimental validation. One simply cannot use a pure analytical method in this complex field with experimental validation.

The primary types of validation have been carried out with experiments that are fairly extreme and carried out in facilities such as LOFT or the Semi-Scale Facility and in the Power Burst Facility. One is left between the two areas, limited experimental validation involving transients which are important but not so extreme.

We would propose then that NRC's code development program move toward that of the industry, utilizing the industry codes, as I mentioned in my testimony, so that we are not faced with using computer codes which provide a very significant overconservatism.

Mrs. **Bouquard**. Are you concerned with the scientific basis of the computer codes?

Dr. **Taylor**. Yes. Again, computer codes which attempt to describe these complex and multiparametric phenomena must have an experimental basis for them. That experimental basis is generally very hard to come by.

For example in thermal hydraulic programs, in looking at extreme transients, experiments are available which provide adequate bases for slug flow occurring in that thermal hydraulic condition but not the laminar flow which exists before slug flow develops in the transition between—which is also very important. That area of validation between what we might call a normal transient that you can see during operation of a powerplant, and that which is performed in experimental facilities whose purpose is to have extreme transients, that area scientifically is still fairly weak.

Mrs. BOUQUARD. Would you explain how the use of the MARCH computer code leads to experiments that are needlessly expensive and not really relevant to reactor safety?

Dr. TAYLOR. Yes. This is of particular importance in the severe accident research and development program plan.

The March computer code is currently the analytical method which is used to describe severe accident conditions. It is because of the very complex technology involved and, again, the lack of thorough experimental validation it entails substantial uncertainties. For example, it does not treat substantial heat sinks which exist in the system in the form of a superstructure which exists above the core.

As a result it predicts, unrealistically high temperatures. As we see the evolution of the research and development planning program, those who have been involved in defining the program have taken the March computer code, computed the entire temperature range, including these unrealistically high temperatures and then proposed the carrying out of a series of experiments at these very high temperatures. We believe such experiments are fundamentally meaningless to the decision that must be made in addressing a severe accident program. The experiments should be devised to establish a benchmark in the more real range of conditions against which such codes can be validated.

Mrs. BOUQUARD. Very good. That was a very good answer.

What technical difficulties do you think need to be addressed in applying the NRC's results on scale-model containment failure modes that you mentioned to full-scale containments?

Dr. TAYLOR. This is a fairly common difficulty in any scaling up process technically in that one cannot necessarily distinguish the relative contributions of different physical features in the system in scaling up. Specifically in the containment case we are not sure whether you can simultaneously scale the effect of concrete aggregate as well as the rebar reinforcement from the scale models all the way to the full-scale containments.

The engineering methods have applied extremely conservative approaches to assure they cover this uncertainty. We judge that there is too much conservatism and we feel that additional experiments to investigate the scaling effect would be required to tie that down.

Mrs. BOUQUARD. Has EPRI conducted any research into the impact of safety related issues on adding a high point vent to existing reactors?

Dr. TAYLOR. We have carried out research on the magnitude of noncondensibles in primary water systems, not specific research on the adequacy and effectiveness of high point vents.

Our experience from the work we have done would simply indicate that high point vents in the entire system should be clearly provided to assure that the favorable impact on safety of a high point vent can clearly be established. Specifically in some systems the steam generators are at a higher point than the reactor vessel and so a simple high point vent in the reactor vessel may not, depending on the circumstances, fully resolve the issue.

Mrs. BOUQUARD. So, you are concerned about the relative impacts?

Dr. TAYLOR. I would say generally the high point vent capability is favorable to safety. There are some special cases that should be examined and I am sure are being examined to assure that that high point vent really applies for the entire primary system.

Mrs. BOUQUARD. Would you relate your concerns in regard to the safety impact of the water level indicators that have been required by NRC in response to the TMI accident?

Dr. TAYLOR. I do not think there is any question that if an unambiguous measurement of the water level in the reactor vessel is provided to the operator this would be an improvement in safety. It is extremely important that as the development and utilization of such water level indicators is pursued that assurance is provided through test and careful introduction, that one truly has an unambiguous measure of water-level indication.

Mrs. BOUQUARD. Is EPRI doing any research in this area?

Dr. TAYLOR. Our research here has been limited to evaluation of devices which have been developed and offered to the utilities by the industry so as to provide guidance to the utilities in their selection and utilization of that type of equipment.

Mrs. BOUQUARD. Thank you very much for your very fine testimony.

We appreciate your being with us today and we look forward to having you before our committee again. Thank you very much, Dr. Taylor.

Dr. TAYLOR. Thank you, Madam Chairwoman.

Mrs. BOUQUARD. Our next witness is Dr. Herbert Kouts. If you will proceed to the witness table, we look forward to your testimony.

The full text of your prepared testimony will be included in the record, and you may summarize if you wish to do so.

STATEMENT OF DR. HERBERT J. C. KOUTS, VICE CHAIRMAN, REACTOR SAFETY RESEARCH REVIEW GROUP, NUCLEAR SAFETY OVERSIGHT COMMITTEE

Dr. KOUTS. Madam Chairwoman, Professor Norman Rasmussen and I were requested to discuss the report of the reactor safety research review group before your subcommittee. Professor Rasmussen was unable to be here today and asked me to convey his regrets. There are not many things which take precedence over the Congress of the United States but it seems that giving a final examination to your course at MIT is one of those things.

Mrs. BOUQUARD. I certainly would concur with his feelings. He certainly has made many valuable contributions to the work of this committee and we are happy to have you here and I understand that Dr. Rasmussen worked with you in preparing the testimony today.

Dr. KOUTS. Yes he did.

Mrs. BOUQUARD. Please proceed.

Dr. KOUTS. Early in 1981 the President's Nuclear Oversight Committee formed this safety research review group to undertake a review of the NRC's safety research program. The group has membership of a list of people in my testimony and I will not repeat it here.

The group spent the spring and summer of 1981 interviewing individuals from both Government and industry. It then met for a week in August of 1981 and formulated a number of conclusions about the strengths and weaknesses of our national program of research in light water reactor safety. The results of this analysis were submitted to the President's Nuclear Safety Oversight Committee in September of 1981, last year.

In considering the conclusions of this group, it is important that you realize the scope of its review. The review was limited to research in the area of light water reactor safety and specifically did not include work in safety of fast reactors.

Furthermore, research was not considered in areas of the fuel cycle other than in the reactor itself, and the generation of electric power. The research programs sponsored by the Department of Energy and by the nuclear industry were included in the review too so that we could judge how well the total national program was integrated.

The total national program in this area in 1981 was conducted at a level of about $300 million. Approximately one-third of this was expended by the private sector and the rest by the Federal Government.

The vast majority of the Federal spending was by the Nuclear Regulatory Commission. Although Public Law 96-567 calls for a considerable DOE program in this area, to date it has not been funded.

The review group reached a number of specific conclusions that will be discussed later. First we would like to comment on some of our more general observations.

The national research program on safety of light water reactors began to assume its present form and objectives only about 8 years ago. The structure it assumed then was in response to questions raised during the long hearings of 1971-73 on the criteria for effectiveness of emergency core cooling systems.

The program has, for the most part, kept this initial orientation ever since with little redirection to topics not originally included. This is understandable in the context of the length of time that has been found necessary to mount such a complex research program and to carry it to useful conclusions.

In the intervening period, the program has made significant progress in answering questions on reactor safety. The NRC, its contractors, and those in industry who have joined in this program have earned and well deserve a substantial measure of credit for important contributions to the safety of the public.

The program has been successful in many ways so that today there is a much higher level of confidence throughout the technical community in the overall safety of nuclear power plants than existed a decade ago, and it only takes memory to substantiate that.

It is, of course, inevitable that in looking back on such a large program a number of ways by which it might be improved can be found. Many of the objectives have now been achieved, and it is now appropriate to carefully examine what could be done.

One of the inevitable questions that arises at such a time is whether or not any more safety research is needed. If in fact a high level of reactor safety has been achieved could we not greatly reduce our spending in this area?

Although on the surface this argument seems logical, the review group felt that this would not be in the best interest of our society for several reasons.

First, as we build more and more reactors it will be necessary to achieve a higher level of safety in each plant if the overall risk to our citizens is to remain about constant. It is this logic that has led to a substantial level of aviation safety long after airplanes were accepted as a safe way to travel.

The second reason is that the safety requirements for reactors include very substantial conservatisms to account for a lack of precision in our knowledge of certain parts of the safety analysis. The only basis on which to reduce these conservatisms is the development of more precise knowledge through research and experience. With this knowledge, reactor costs could be reduced and our uranium resources could be used more efficiently. Even very modest reductions in cost for modest improvements in efficiency would quickly pay back the research investment.

The review group feels that an adequately funded, well managed research program could pay back its investment many times over. For this to be true, it is essential that the results of the research be integrated into the plant safety requirements.

It may be asked why the Nuclear Regulatory Commission should be involved at all in research to reduce conservatisms. There are several reasons.

First, it is important that conservatism be confirmed, and the ability to reduce it is only a by-product of evaluating it.

Second, the conservatisms that have been applied are a social and economic penalty on society, resulting from inadequate knowledge at the time they are applied. It is important that regulatory bodies not penalize society any more than is called for by prudent regard for safety.

Now I would like to turn to some of the specific conclusions of the research group. These conclusions were developed in the extensive report which was furnished to the President's Committee, and which has been given this subcommittee for the record.

A subset of recommendations was extracted as more important than the others and I would like to briefly go over these if you will allow it.

Mrs. **Bouquard**. Please proceed.

Dr. **Kouts**. Thank you, Madam Chairwoman.

Our first recommendation was that the LOFT program at the Idaho National Engineering Laboratory should be phased out in an orderly manner. We were divided as to whether the program should be completed in fiscal year 1982 or 1983. This recommendation preceded the present negotiations for multinational sponsorship and funding of LOFT and that recommendation might have been different if these negotiations bear fruit.

We then recommended a number of new programs that should be uncertaken by NRC or DOE or by the industry as appropriate. Some of these could be funded from sources made available by phasing out of LOFT or slowing it down.

We recommended that a major effort be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems for both the reactor vessel and the containment.

We recommended that a program be initiated to develop and evaluate methods of providing more reliable station electric power, both a.c. and d.c.

We recommended that a program be undertaken to examine the relative safety merits of rigid versus flexible design of piping. We were uneasy about the wisdom of the rigid piping designs incorporated in plants to withstand earthquakes, partly because these designs do not recognize the ability of piping to deform without breaking. Too many pipe hangers and shock absorbers can introduce a penalty, making breakage more possible.

We recommended industry or DOE programs to develop valves that would be more reliable than present ones. Valves are considered to be among the least reliable components of nuclear plants. A sticking relief valve caused the trouble at Three Mile Island.

We suggested that DOE and industry should proceed with programs on decontaminating the primary systems of nuclear plants. This is important in reducing exposure of operating staff personnel.

We suggested a joint DOE and NRC program to develop faster and more extensive simulator hardware and software, to analyze plant transients. This would improve training of operators and would also improve the ability to survey accident scenarios in a parametric way. We recommended research programs to resolve concerns as to water hammer, where rapid condensation of steam and water causes pressure surges that could in principle damage piping and circuit components.

As a last area of nuclear research we recommended increased attention to the problem of sabotage.

We also made a number of recommendations concerning research already underway. The more important ones of these were the following:

We recommended a substantial increase in research bearing on the man-machine interface, the area of so-called human factors. In particular, we mentioned control room design, operator aids, and selection and training of operators.

We recommended research to improve understanding of quantities of radioactive materials that might be released to the environs in a severe accident; this is the source term research program. There is now widespread belief that this source term may have been greatly overestimated in the past.

We believe that research on small, loss-of-coolant accidents had not yet led to definitive results, and it should be continued. This is apart from the large loss-of-coolant accident research connected with the LOFT program.

We were concerned as to the possibility and effects of repressurizing reactor vessels in the cold state and recommended programs to insure early resolution of outstanding questions.

We believe that steps should be taken to insure Federal funding of the recovery of the damaged reactor core at Three Mile Island No. 2. This reactor core is a potential treasure of valuable reactor safety information which must be taken advantage of.

We recommended further research on the possible evolution and burning of hydrogen gas in a severe reactor accident.

We believed there should be extended use of probabilistic analysis in a number of areas which we discussed.

We recommended increased research on plant control systems and research to improve readiness for several hearings that were contemplated at the time, especially hearings on minimum-engineered safeguards features of nuclear plants and the proposed degraded core rulemaking.

Those were our recommendations as to specific research programs that were ongoing. We had some important general recommendations as to NRC research programs. We were concerned that the results of NRC's research programs do not seem to be used in nuclear reactor regulation. We recommended that the Commission encourage and promote the visible integration of the results of research into the regulatory process.

We were opposed to NRC policies that tied the research programs to confirmatory objectives and suggested that research should be allowed to point the way to improved safety when this seemed appropriate.

We had several suggestions concerning structuring of the NRC's long-term research program including especially the view that the five Commissioners should begin to introduce and continue guidance of this program at the Commission level because this guidance has been lacking up to the present.

We suggested that risk assessment to used as one tool in setting research priorities.

Finally, we were concerned that the Department of Energy has not assumed the responsibilities for reactor safety research that were required by Public Law 96-567, especially because the necessary funding never became available.

We recommended that DOE vigorously enter the field of reactor safety research, and that DOE improve the strength of staff needed to plan and manage this research.

In summary, Madam Chairwoman, the report of the reactor safety research review group contained many recommendations that are for the most part still valid and that we believe could substantially enhance nuclear power plant safety.

The report had the misfortune to appear at a time of transition in politics, and so did not receive the scrutiny that it would more normally would have had.

Few of the recommendations have had much impact up to this point and we are naturally disappointed. We recommend that our report be taken out, dusted off, and considered seriously.

[The prepared statement of Dr. Kouts follows:]

Testimony By

Dr. Norman C. Rasmussen
and
Dr. Herbert Kouts

Mr. Chairman, we have been asked to discuss the Report of the Reactor Safety Research Review Group. Early in 1981 The President's Nuclear Safety Oversight Committee (NSOC) formed this group to undertake a review of the Nuclear Regulatory Commission's research program in the area of nuclear reactor safety. The group had the following members:

Dr. Norman C. Rasmussen, M.I.T., Chairman
Dr. Herbert J.C. Kouts, Brookhaven National Laboratory, Vice Chairman
Dr. Spencer Bush, Battelle Pacific Northwest Laboratory
Dr. Thomas Connolly, Stanford University
Dr. Herbert G. MacPherson, Institute for Energy Analysis
Dr. David Okrent, University of California at Los Angeles
Mr. Lombard Squires (retired), E.I. duPont de Nemours Company
Dr. Edwin Zebroski, Nuclear Safety Analysis Center

The group spent the spring and summer of 1981 interviewing individuals from both government and industry. It then met for a week in August of 1981 and formulated a number of conclusions about the strengths and weaknesses of our national program of research in light water reactor safety. The results of this analysis were submitted to The President's Nuclear Safety Oversight Committee in September of 1981.

In considering the conclusions of this group, it is important that you realize the scope of its review. The review was limited to research in the area of light water reactor safety, and specifically did not include work in fast reactor safety. Further, research was not considered in areas of the fuel cycle other than the reactor itself. The research programs sponsored by the Department of Energy and the nuclear industry were included, so we could judge how well the total national program was integrated.

The total national program in this area in 1981 was conducted at a level of about 300 million dollars. Approximately one-third of this was expended by the private sector and the rest by the federal government.

The vast majority of the federal spending was by the Nuclear Regulatory Commission. Although Public Law 96-567 calls for a considerable DOE program in this area, to date it has not been funded.

The review group reached a number of specific conclusions that will be discussed later, but first we would like to comment upon some of our more general observations.

The national research program on safety of light water reactors began to assume its present form and objectives only about eight years ago. The structure it assumed then was a reaction to questions raised during the long hearing of 1971-73 on the criteria for effectiveness of emergency core cooling systems. The program has, for the most part, kept this initial orientation ever since, with little redirection to topics not originally included. This is understandable in the context of the length of time that has been found necessary to mount such a complex research program and to carry it to useful conclusions. In the intervening period, the program has made significant progress in answering questions on reactor safety. The NRC, its contractors, and those in industry who have joined in this program have earned and well deserve a substantial measure of credit for important contributions to the safety of the public.

The program has been successful in many ways so that today there is a much higher level of confidence throughout the technical community in the overall safety of nuclear power plants than existed a decade ago. It is, of course, inevitable that, in looking back of such a large program, a number of ways by which it might be improved can be found. In addition, by now many of the original objectives have been achieved, so that it is now appropriate to carefully examine just how the program should be reoriented.

One of the inevitable questions that arises at such a time is whether or not any more safety research is needed. If in fact a high level of reactor safety has been achieved couldn't we greatly reduce our spending in this area? Although on the surface this argument seems logical, the review group felt that this would not be in the best interest of our society for several reasons. First, as we build more and more reactors it will be necessary to achieve a higher level of safety in each plant if the overall risk to our citizens is to remain about constant. It is this logic that has led to a substantial level of aviation safety research long after airplanes were accepted as a safe way to travel. The second reason is that the safety requirements for reactors include very substantial conservatisms to account for the lack of precision in our knowledge of certain parts of the safety analysis. The only basis on which to reduce these conservatisms is development of more precise knowledge through research and experience. With this knowledge, reactor costs could be reduced and our uranium resources could be used more efficiently. Even very modest reductions in cost or modest improvements in efficiency would quickly pay back the research investment. The review group feels that an adequately funded, well managed research program could pay back its investment many times over. For this to be true it is essential that the results of the research be integrated into the plant safety requirements.

It may be asked why the Nuclear Regulatory Commission should be involved in research to reduce conservatisms. There are several reasons. First, it is important that conservatism be confirmed, and the ability to reduce it is only a by-product of evaluating it. Second, the conservatisms that have been applied are a social and economic penalty on society, resulting from inadequate knowledge at the time. It is important that regulatory bodies not penalize society any more than is called for by prudent regard for safety.

Now let us turn our attention to the specific conclusions of the review group. These conclusions are developed in the extensive report which was furnished to The President's Nuclear Safety Oversight Committee, and which has been given this Subcommittee for the record. A subset of recommendations was extracted as more important than the others, and we should like to take this opportunity to discuss these.

Our first recommendation was that the LOFT program at the Idaho National Engineering Laboratory should be phased out in an orderly manner. We were divided as to whether the program should be completed in FY-1982 or FY-1983. This recommendation preceded the present negotations for multinational sponsorship and funding of LOFT, and might be different if these negotiations bear fruit.

We then recommended a number of new programs that should be undertaken by NRC or DOE or the nuclear industry, as appropriate. Some of these could be funded from sources made availble by reduction or phasing out of the LOFT program.

We recommended that a major effort be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems, for both the reactor vessel and the containment. Though systems in use have a reliability that has been regarded as satisfactory, we believed that improvements were both possible and worthwhile.

We recommended that a program be initiated to develop and evaluate methods of providing more reliable station electric power, both AC and DC. This recommendation was made because one possible initiator of a large accident is the conceivable loss of all electric power.

We recommended that a program be undertaken to examine the relative safety merits of rigid versus flexible design of piping. We were uneasy about the wisdom of the rigid piping designs incorporated in plants to withstand earthquakes, partly because these designs do not recognize the ability of piping to deform without breaking. Too many pipe hangers and shock absorbers can introduce a penalty, making breakage more possible. A design study is badly needed.

We recommended industry or DOE programs to develop valves that would be more reliable than present ones. Valves are considered to be among the less reliable components of nuclear plants. A sticking relief valve caused the trouble at Three Mile Island.

We suggested that DOE and industry should proceed with programs on decontaminating the primary systems of nuclear plants. This would reduce exposures of operating staff personnel.

We suggested a joint DOE and NRC program to develop faster and more extensive simulator hardware and software, to analyze plant transients. This ability would improve training of operators, and also the ability to survey accident scenarios parametrically.

We recommended research programs to resolve concerns as to water hammer, where rapid condensation of steam in contact with water causes pressure surges that could in principle damage piping and circuit components. Water hammer is a phenomenon common to all kinds of steam systems.

As a last area of new research, we recommended increased attention to the possibility of sabotage.

We also made a number of recommendations concerning research aleady being done. The more important comments on existing research were as follows:

We recommended a substantial increase in research bearing on the man-machine interface, the area of so-called human factors. In particular, we mentioned control room design, operator aids,and selection and training of operators.

We recommended research to improve understanding of the quantities of radio active materials that might be released to the environs in a severe accident. There is now widespread belief that this may have been greatly overestimated in the past.

We believed that research on small loss-of-coolant accidents had not yet led to definitive results, and it should be continued.

We were concerned as to the possibility and effects of repressurizing reactor vessels in the cold state, and recommended programs to ensure early resolution of outstanding questions.

We believed that steps should be taken to ensure federal funding of the recovery of the damaged reactor core at TMI-2. This reactor core is a potential treasure of valuable reactor safety information, which must be taken advantage of.

We recommended further research on the possible evolution and burning of hydrogen gas in a severe reactor accident.

We believed there should be extended use of probabilistic analysis in a number of areas.

We recommended increased research on plant control systems.

We recommended research to improve readiness for several hearings that were contemplated at the time, including hearings on minimum engineered safeguards features of nuclear plants, and the proposed degraded core rulemaking.

These were our recommendations as to specific research programs that were ongoing. We also had several important general recommendations as to NRC research programs.

We were concerned that the results of NRC's research programs do not seem to be used in nuclear reactor regulation. We recommended that the Commission encourage and promote the visible integration of the results of research into the regulatory process.

We were opposed to NRC policies that tied the research programs to confirmatory objectives, and suggested that the research should be allowed to point the way to improved safety when this seemed appropriate.

We had several suggestions concerning structuring of the NRC's long-term research plan, including especially the view that the five Commissioners should introduce and continue guidance at the Commission level.

And we suggested that risk assessment be used as one tool in setting research priorities.

Finally, we were concerned that the Department of Energy has not assumed the responsibilities for reactor safety research that were required by Public Law 96-567, partly because the necessary funding never became available. We recommended that DOE vigorously enter the field of reactor safety research, and that DOE improve the strength of the staff needed to plan and manage this research.

In summary, Mr. Chairman, the report of the Reactor Safety Research Review Group contained many recommendations that are for the most part still valid, and that we believe could substantially enhance nuclear power plant safety. The report had the misfortune to appear at a time of transition in politics, and so did not receive the scrutiny that it would more normally have had. Few of the recommendations have had much impact up to this point, and we are naturally disappointed. We recommend that our report be taken out, dusted off, and considered seriously.

Mrs. BOUQUARD. Thank you very much for your testimony. I think you raised a very interesting point in your testimony about the relationship of conservatism in the regulations to reactor cost and efficient utilization of our uranium resources. You felt that conservatism can be reduced through more precise knowledge from research and experience—that conservatism that has been applied is a social and economic penalty on society resulting from inadequate knowledge at the time. This is certainly a different viewpoint. Would you like to elaborate more on this?

Dr. KOUTS. Yes, Madam Chairwoman. I would be happy to do that.

In the course of regulating for the safety of the public, it is frequently the case that one runs up against a technical problem for which the complete solution is not known. In this case it is necessary sometimes to assume, not necessarily the absolute worst but enough so that——

Mrs. BOUQUARD. The worst possible case.

Dr. KOUTS. Yes, a sort of worst possible case.

Mrs. BOUQUARD. Or a responsible scenario.

Dr. KOUTS. If you do that, you commonly impose a penalty on society. In this particular case, you probably make electricity cost a lot more and you probably cause a lot more plants to be built than are absolutely necessary because you cut into the capacity factor of plants by various measures that you take.

If this has to be done, we believe that it is imperative that whatever body is forced to take these measures also assumes the responsibility for minimizing them in the long run by establishing a basis for more rational solutions of these problems.

Mrs. **Bouquard**. That is very interesting. What sort of research—you mentioned further funding for research at TMI. What research do you believe ought to be conducted after the TMI-2 core is removed?

Dr. **Kouts**. Madam Chairwoman, it is necessary that we find out to what extent damage was sustained by that core and what the physical and chemical processes were. It is necessary for us to find out what happened to the control elements in this reactor. It is necessary for us to find out to what extent the various things to which this core was subjected played a role in the final damage that was sustained. It is necessary for us to find out where all of the fission products went and in what chemical and physical form. This ties in with source term conservatism that I mentioned earlier.

If I may turn to the source term question that is related to this particular point, a great deal of attention is given in discussions of the source term to the fact that not very much iodine was released in the course of this accident, and that is a remarkable and very important observation. An even more important observation, it seems to me, is: What has happened to the cesium, strontium, and rubidium, which are other important elements in the analysis of safety of plants of this kind. Less of these were released than of iodine. This has historically been the case in other instances where the core has been damaged in one way or another.

The conservatism in these particular areas is retained at an even higher level, I believe, than in the case of iodine. It is important to resolve that question.

Mrs. **Bouquard**. This is a very significant question. As you know, members of this committee have objected to research and development funding spent at TMI and I wanted your perspective on that.

I noticed also that the structure of your report differed very significantly from NRC's long-range research plan that was revealed to us yesterday. Would you give us a little more detail about what led you to structure your comments as you did starting with your emphasis on conservatism from a different viewpoint?

Dr. **Kouts**. One of our recommendations actually dealt with the structure of NRC's long-term plan—the way their long-term plan is put together.

We recommended that instead of having it divided by the decision units, as it now, that it be structured according to organic interests, to the problems which NRC wishes to solve. It is now structured more according to the way NRC is put together than according to what problems they are trying to solve.

We suggested that if they were to restructure the plan, the work they do might take on a different structure, too.

Mrs. **Bouquard**. Do you also feel that the Commission should have more interaction in looking into projects?

Dr. **Kouts**. The Nuclear Regulatory Commission has not given very much guidance to the research program. It has not told where it thinks its major problems are or where it thinks major problems in the future will be found and if it could begin to give guidance of this kind I think it would have a very important impact on the research program.

Mrs. **Bouquard**. You referred to the small loss-of-coolant accidents that you would like to see more work and funding done on. What kind of results would you like to see come out of this research?

Dr. **Kouts**. The problem that we have now is that the computer codes, which were discussed in some detail by Mr. Taylor a few minutes ago, have really not been tested adequately in the areas of phenomenology which would appear in small loss-of-coolant accidents.

A few tests of larger size than have been conducted in this area until now would be very beneficial in testing these computer codes. We retain a great measure of conservatism in these areas which we could probably clear up.

May I mention one area in that connection?

Mrs. **Bouquard**. Please go ahead.

Dr. **Kouts**. All of these things tend to overlap and influence each other. There is a great deal of interest these days in the question of pressurized thermal shock and this has been made almost a public issue rather than a technical issue in safety.

One of the important questions in pressurized thermal shock, which is what would happen if you suddenly injected cold water into a reactor vessel which has been running at a very high temperature, is to what extent does the water, in its passage in, mix with the water that is already there. You get temperature of some mixture of the water rather than the temperature of the cold water itself. The extent of thermal shock is going to be tied to how this mixing phenomenon takes place. We do not fully understand this problem at present. This is one of the open areas of the small-break analysis or small-break loss-of-coolant accident. This is where areas interact with each other.

Mrs. **Bouquard**. One of the messages that I sort of got out of your testimony was that I feel that you are looking for applied research. In that vein I would like you to be a little more specific on your statement that NRC research results are not fully reflected in the regulations. Would you give us a few examples?

Dr. **Kouts**. Yes, Madam. The NRC approach to the loss-of-coolant accident at present is set by what is called appendix K to part 50 of section 10 of the Code of Federal Regulations. Appendix K provides for a very prescriptive process, by which it is necessary for licensees or applicants for licenses to analyze how their plants would respond to possible loss of coolant accidents.

The way by which this analysis is done is very carefully laid out and contains many of these conservatisms which I have mentioned earlier.

The course of the safety research program over the years has shed a great deal of light on some of the more important of these areas. One of these is the amount of heat which is generated by fission products following shutdown of the reactor.

At the time appendix K was written it was necessary to take a somewhat conservative view of how much heat is generated because there was still uncertainty. One of the important contributions of the program has been to show how big this conservatism is. That has not been introduced into the licensing process by any means although it is generally agreed that changing the fission product source-term by the amount which the experimental program has shown to be available would have a substantial effect on the outcome of those calculations.

There are similar results connected with the rate of oxidation, and the amount of heat which might be given up by zircaloy cladding of the fuel if it were damaged. These have been conservatively overestimated in the appendix K rule in the past, and we believe that they could now be substantially relaxed with beneficial results.

Mrs. **Bouquard**. Thank you very much. We appreciate your testimony.

We will stand in recess for another vote and we will resume our hearings shortly. Thank you very much.

[Whereupon the subcommittee stood in recess at 3:25 p.m.]

Mrs. **Bouquard**. We will resume our hearings.

Our next witness is Dr. Edwin L. Zebroski who is vice president of engineering, Institute of Nuclear Power Operators.

We cannot say much about the continuity of our hearings today but I certainly must say that the quality has been excellent and it certainly surpasses the contribution of the Congress of the United States today.

Please proceed with your testimony. The full text of your written testimony will be made a part of the record and you may summarize if you wish to do so.

STATEMENT OF DR. EDWIN L. ZEBROSKI, VICE PRESIDENT, ENGINEERING, INSTITUTE OF NUCLEAR POWER OPERATORS

Dr. **Zebroski**. Thank you, Madam Chairwoman.

I appreciate the opportunity to comment on some elements of the NRC's research and development planning and efforts. I would also take the opportunity to summarize somewhat the written comments.

Mrs. **Bouquard**. Please proceed as you wish, Dr. Zebroski. Your entire statement has been made a part of the record.

Dr. **Zebroski**. The background of the broad commentaries by Dr. Kouts and Dr. Taylor makes it possible for me to focus on a few selected elements of the program which have the most bearing on the interests of INPO and its industry supporters.

The elements that I will discuss are mainly in sections 7 and 10 of the draft NUREG-0784. If I might address several of the activities which are under way at the Institute of Nuclear Power Operations are related in several ways to portions of the NRC program. INPO does no research as such, but a considerable part of our analytical work and of the application engineering work either parallels or bears directly on NRC work. Many of these efforts are complementary and have good communication and coordination with the NRC.

INPO is also a user of the practical results of research from both Government and industry. One of our functions is to help find, to recommend, and to define best practices for utilities. These involve the application of best available technology, including methods and management systems some of which are a result of R&D.

Conversely, the planning and the goal setting for the research and development in the NRC may derive some useful inputs and perceptions from the several kinds of activities at INPO. I will highlight just four of these:

We do comprehensive reviews of all U.S. plant operating experience. I mean literally all. We not only do the LER's but also plant incident reports. We are now also looking at human error reports from a number of utilities on a pilot basis. This operating information is now supplemented by added information which is beginning to flow from utilities in 12 foreign countries. There are 22 foreign utilities that are also participants in INPO and who are tied together with us via a real time satellite communication system, plant to plant—which is now virtually worldwide.

The member utilities overseas now represent more a bigger total number of operating reactors than we have in the United States, so this is an increasingly important source of practical information and experience.

Second, we do intensive analysis of all this operating experience and we try to derive practical corrective actions from these analyses. Corrective actions are sought and recommended which can reduce the probability of future troublesome occurrences, or reduce their consequences if they do occur.

From over the 8,000 events that we have analyzed in the last 2 years we have sifted out approximately 180 of these which are significant enough to warrant recommendations and actions for remedies.

Third, we have a system for prioritization of the safety importance of the events which are analyzed. If we gave 8,000 undifferentiated reports to our member utilities that would be worse than useless. The sifting down to a level where approximately one significant message per month reaches a given unit helps to focus attention on it. These priorities are based on our relative assessment of safety impacts and based on the potential or plausible impacts of such events plus likely variations of them. This bears on many of the questions which have to do with prioritization of safety research and development by similar assessment processes.

Finally, the fourth item is that we have a systematic and continuing evaluation of plant operating practices and now also starting to include management practices which are determined by plant evaluation visits. We send balanced teams of people who are highly experienced in plant management operations, maintenance, and technical support. We have evaluated, reported, and negotiated remedies with every operating power unit in the United States and are now the second cycle of such visits. The cycle is approximately 15 months at present for full turnaround full evaluation visits.

INPO has mounted a new effort which is now in the pilot stage, for evaluation of the construction quality practices and design control practices for plants under construction. We can report further on that effort when it matures in September.

I would like to digress and talk about the issue of confirmatory research versus problem resolving research and development which several previous speakers and Dr. Siess yesterday have alluded to in their testimony.

The research and development program of NRC may be regarded as involving these two types of results: those related to "assist in the establishing of regulations for existing or future facilities (quoted from NUREG-0885, the current NRC policy document and second, the problem solving work related to providing information or data which can help exploit an opportunity for practical improvements which can help to or solve well-defined problems in designor in operation.

The objectives and results from confirmatory research reflect the perceived needs and goals of the regulatory staff as noted by Dr. Siess in his testimony yesterday. Comments on this aspect of the research and development plan are very difficult since they involve value judgments on the goals result or means of regulations. My comments herein, therefore, apply principally to those aspects of the research and development program which we perceive as being related to problem solving in the sense noted above and in which many cases parallel problem solving efforts in the industry.

At the time the NRC was separated from the former Atomic Energy Commission, the NRC program was directed by the 1974 legislation to restrict itself to confirmatory research. This policy in effect still controls most of the formulation of the current research and development program.

For example, the 1982 policy guidance document from the Commission's NUREG-0885 states, "the purpose of the research program is to assist in establishing regulations." This question of policy, I believe, underlies the unease evident in many of the reviews and comments on the NRC program in recent years. This includes those which Dr. Kouts summarized from the NSOC working group—in which I also participated. Within the constraints of this legislation, the NRC staff has become sensitive, at least in principle, to the limitations of this pattern, for example the extreme emphasis on highly improbable or hypothetical situations still is a major part of the R&D program but no longer completely dominates it.

There is an increased recognition that the "limiting case approach not only may lead to unwarranted conservatism but may result in design requirements that inhibit the ability to cope with actual accidents or precursors of potential accidents."

Mr. Chipman referred to several examples of this kind earlier. I believe this is a valid and very important perception, I agree with his comment that in the area of the detailed content of the program (the expected useful results) it is a little disappointing that the general philosophy which is stated in the introduction is not fully reflected in the planning.

R&D and regulation which considers mainly limiting cases shows up as a difficulty when we study actual operating events in plants. We often need to evaluate the operations done by the operator in the actual situation present. He finds added difficulty in operation whenever the procedures, remedies, and operating training are based on assumed, hypothetical, worst-case situations, which he never actually sees.

The staff plan recognizes these issues in philosophy by six important statements which are in NUREG-0784. I will mention three of these because they bear directly on work that we are involved in at INPO.

The first one is the stated objective of setting better understanding of system response and safety margin provided by plants in real-world accidents and transients. That is the point of going to the actual experience versus the hypothetical. I endorse this strongly.

Second, the analysis and appreciation of needed operational capabilities, operator training requirements, and human factors. The bulk of my remaining comments will be in this area.

Finally, there is a third item on using probabilistic risk assessment to give a better ability to differentiate the relative safety significance of different regulatory issues or prospective requirements. I think everyone will encourage the need for this.

Five of the six topics listed in the NRC general objectives I think are laudable and generally warranted. I share the misgivings expressed by Dr. Kouts and Dr. Taylor on the emphasis on mitigation systems. Perhaps a little caricature of this is to say that a great deal of effort aimed at finding how bad things are when the worst happens is perhaps less productive than finding out ways to make sure that those things are made so unlikely that they never will happen. I think that the public interest might prefer more effort on prevention rather than mitigation.[1]

Of the three items, the most interesting one (because it has such broad application to so many aspects of the program) is the area of human factors. We are involved in that aspect and also the operator training and the man/machine interface.

We work as technical staff to support the substantial industry working groups on this topic. There are working groups, owners groups, committees, and development activities from the various contractors. Our efforts are mainly to help make the best application of technology which is largely known, based on recent years of research at EPRI in the human factors' area. There are also useful results and many publications from the aircraft industry, from the space program, and from military programs. These have been involved in the ergonomics or human factors' area for perhaps at least 15 years in ways that are very often directly applicable to the nuclear program.

Our emphasis is to give appropriate attention to those applications of human engineering which have the most direct bearing on reducing the known elements of small and medium-sized accidents. We know this from the actual analysis of a great many such events from the operating experience.

Important as that is, however, we give the main priority to preventing or recovering successfully from severe accidents or accidents with potential for very large consequences.

Of the many questions on the prioritization process my personal opinion is that this represents the most direct and productive way to get the most constructive output from R&D. That is, to prioritize the work according to the expected impact of the results on the risk avoided. I think that Dr. Taylor, Dr. Kouts, and Mr. Chipman all made a similar point.

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[1] Note added in draft: Mitigation can merge logically with "prevention," in the sense of "prevention of severe consequences" if "mitigation" is interpreted to include recovery from serve situations even if the initiating events were not prevented.

Coming to the question of probabilistic risk assessment, having been a practitioner of this myself I would say that I am skeptical of many of the proposed applications.

There is one application which is clearly very important, very practical, and very usable and that is relative risk assessment of alternate options.

For example, if you have a proposed problem or a known problem you always have alternate procedures or proposed changes to consider. There are usually straightforward ways now of making that relative assessment with great validity.

The relative assessment can also be used to set priority and timeliness targets for implemention of remedies in plants. It is also a powerful technique (as mentioned by the ACRS on several occasions) for setting priorities in research and development.

I however do agree with the many concerns which have been expressed on the use of global risk assessments——

Mrs. BOUQUARD. Excuse me, Dr. Zebroski. We are going to have to break one more time. We will stand in recess for about 5 minutes.

[Whereupon the subcommittee stood in recess at 4:04 p.m.]

Mrs. BOUQUARD. We will resume our hearings. Please proceed, Dr. Zebroski.

Dr. ZEBROSKI. Thank you. I was talking about probabilistic risk assessment.

We use this relative risk assessment approach to prioritize significant operating experience reports which are provided to all U.S. nuclear utilities and now those in the 12 countries overseas. These include recommendations for remedies. We cover not only events which are obviously troublesome, but also many which are benign in the actual case, but which have plausible variations which could become more serious.

Another use for the risk assessment technique is to look at low-probability, high-consequence accidents. For this case the technique provides an important estimate that over half of the total risk exposure arises from a specific kind of operator error which can arise in unexpected situations for which event-based procedures or analyses are not practical. I will explain why they are not practical.

The type of error involved is called cognitive error. This means that the operator misjudges the situation he is in for any one of several reasons. He may then execute a set of procedures correctly, thoroughly, and diligently but which are not selected so as to cope fully with the actual situation.

The human factors emphasis in this area is on skill-based or manipulative errors. This is important for some operating events, but I believe it is much less relevant to serious accidents. Unless the manipulative error is compounded by a cognitive error it rarely if ever leads to a serious accident. The reason is that a manipulative error is relatively easy to detect and to correct if you are thinking right. However, if the cognition is not good enough then the error can be compounded.

The reason that specific event-based analyses are impractical for low-probability circumstances in which there may be several failures is because a number of possible combinations is impractically large—it can become near infinite. Accordingly, an approach which uses safe, symptom-based responses which we call emergency operating procedures has been formulated which does not depend on the exact analysis of the sequences.

This approach is alluded to and endorsed in the NRC action plan and is the subject of an intensive effort by all four of the industry owners groups—now with staff support from INPO. We have published documents on the formating and structure of such emergency procedures. The plant-specific, procedures are being generated now for many plants.

Another principal finding on human factors from the TMI accident and others that we have looked at is that the distribution of critical information over a large area of the control room is diluted and confused by the multitude of alarms and the large volume of less important information. The less important information is scattered as to be hard so to perceive under disturbance conditions but is necessary for regular operation. The concentration of the critical plant information into a compact set of displays is now widely perceived as the principal way to be responsive to this evident need. This step is responsive to the major human engineering deficiency also being found in many current reviews of control room designs.

We believe that the way to solve this most important human factor problem requires a complementary triad of developments. They can provide a major gain in the reliability of the operator response in managing rare but potentially serious events. These are:

First, symptom-based emergency operating procedures.

Second, the computer-aided safety displays of key information, the so-called SPDS, safety parameter display system.

Third, requisite training to implement and maintain the continued capability to make effective use of the procedures and this key information.

The industry has come to the perception that the difficulties which have been experienced for the last 3 years in specifying these activities and which have delayed the application of this major risk production option can now be surmounted. The essential perception is that no one of these activities can be useful by itself alone. You must have the triad of procedures, training and displays and prove them out as a matched set. Each one is complementary to the other two and cannot be fully specified except in concert with the other two.

Here again substantial efforts are underway in industry with INPO support and joint INPO/EPRI projects on the safety panel and on the formating and definition of emergency operating procedures. Initially these procedures must be brought into operation in the normal control room by focusing the operator's attention, using training and procedure, on the key information which is already available. Later, this will be supplemented with a matched set of computer-aided displays—the concentrated displays—and the appropriate training modules.

The safety displays also lend themselves to effective, computer-aided training modules and techniques. The approach that I am describing here is now in close agreement with NRC staff positions delineated in a recent staff paper, SECY-82-11.

The research and development plan in the draft document 0784 may benefit by further revision to take account of these developments which, I believe, were largely a cooperative effort by two of the NRC's divisions but perhaps with not enough participation by research.

I would like to make also a comment on the translation of research and development results to practical application since INPO is heavily involved in this area.

It is a common observation that the results of laborious and costly research often find very little actual application. This concern has been raised by NRC research staff and is supported by the review by the NSOC subcommittee covered by Dr. Kouts.

It is true that the tangible results of past research and development—tangible results in the sense of application—are actually rather hard to find. One may still be concerned that this is likely to continue to be the case for the $1 billion, 5-year program as it is now structured. This situation is not the fault of research, per se, nor of regulations.

The most successful results of research will go to application only if you provide a number of additional ingredients. These may seem commonplace but they are nevertheless crucial. The essential ingredients include:

1. A perceived need by those who can apply the results in practice.

2. A way of determining priority for responding to perceived needs, relative to other pressing needs.

3. Access to resources that permit implementation. This depends on the perception of a favorable ratio of benefit gained to the resources required. This perception must be shared by not only NRC but by utility management and by utility commissions.

4. There must be management resolve and capability to marshall the material and technical resources required.

5. We must have a regulatory climate policy and practices which recognize the need for practical tradeoffs at each stage in the formulation, planning, scheduling, and implementation. This must be based on factual assessment of real-world problems in application, often on a plant-specific basis.

I may put in an aside here that there is an important pioneering effort in the NRC, involving the CRGR, which attempts to make these implementation practices more practical. This is connected with the emergency procedures and the safety display panel and we very much hope that that effort proceeds and matures well.

Of those five factors, the work of the industry working groups, EPRI and INPO bear very strongly on the first four elements. The work of the commission staff—and particularly currently of the Committee for Review of Generic Requirements—bears heavily on the fifth requirement which makes implementation possible.

The research and development program plan takes many laudable steps in the direction of addressing real-world problems. I would suggest however that there could be better coupling with the lessons learned from INPO's review of operating experience and the findings and implications of the plant evaluation process. These can help in planning and formulating research and development tasks which have increased likelihood of reaching application.

I might comment also that on the question of the prioritization process, there are two steps to prioritization. One is to decide what are important issues. That is probably relatively easy, to define the playpen in which you should work.

There is a second problem, however, of prioritization which is how do you formulate the targets of expected results in a way that they are most likely to go to application. There I think we have considerably less success. It is a very tough area for planning.

It has been a pleasure to have the opportunity to speak to the subcommittee on this topic. There are a large number of related topics which are also of direct interest to industry and INPO. For example, accreditation of training programs, the criteria and evaluation of operations and management, emergency response planning, radiation control, and health physics.

I have chosen to highlight those few topics which seem to have the largest potential in gains for actual as well as perceived levels of safety. These are also areas which I am happy to say that the NRC and industry efforts are increasingly synergistic, which is to say cooperative and constructive rather than adversarial.

We share with the members of the subcommittee and NRC the common motives of helping to maintain a major element of the energy supply of the United States and to help make it increasingly safe and effective in serving the public.

Thank you.

[The prepared statement of Dr. Zebroski follows:]

STATEMENT TO SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION OF THE HOUSE COMMITTEE ON SCIENCE AND TECHNOLOGY U.S. HOUSE OF REPRESENTATIVES

Edwin L. Zebroski
Vice President
Institute of Nuclear Power Operations
Atlanta, Georgia
May 19, 1982

My name is Edwin L. Zebroski, I am Vice President for the Institute of Nuclear Power Operations and Director of the Analysis and Engineering Division.

I appreciate the opportunity to comment on some elements of the NRC's R&D planning and efforts.

Last fall I participated in the effort of a subcommittee for the Nuclear Safety Oversight Committee which undertook a broad review of the NRC's R&D program. This subcommittee, chaired by Professor Norman Rassmussen, conducted an intensive review and produced a document with comments and recommendations regarding the thrust, objectives, and priorities of many elements of the NRC R&D program as it was then planned. My understanding is that Dr. Herbert Kouts will review the views of this subcommittee in these hearings. As a member of the subcommittee chaired by Professor Rassmussen, I drafted portions of the committee report, and find myself largely in agreement also with those sections drafted by others on the subcommittee.

Also in the hearings of this subcommittee, Mr. John Taylor, Director of the Nuclear Division of Electric Power Research Institute, will provide a broad based review of NRC R&D program. Until last fall, I was Director of the Nuclear Safety Analysis Center at EPRI and former Director of its Systems and Materials Department. I have had the opportunity to read John Taylor's testimony, which also represents an excellent general commentary on the broad program. This background of broad commentaries permits me to focus on a selected few elements of the NRC program for FY 1984. (These elements are mainly in Section 7 and 10 of draft NUREG 0784.)

Activities are underway at the Institute of Nuclear Power Operations which are related in several ways to some portions of the broad NRC program. INPO does no "research" as such, but a considerable part of our analysis work and application engineering support efforts bear directly or parallel NRC research work. Many of these efforts are complementary.

INPO is also a user of useful results of research from both government and industry. One of our functions is to help find and define "best practices" for utilities. These usually involve the application of best available technology, methods, or management systems. Conversely, the planning and goal-setting for NRC R&D may derive useful data and perceptions from the following kinds of activities at INPO:

1. Comprehensive reviews of all U.S. plant operating experience.

   This is supplemented by information beginning to flow from utilities in 12 foreign countries who are participants in INPO. These utilities together represent more operating units than there are in the U.S.

2. Intensive analysis of all operating experience reports, and derivation of corrective actions from this analysis.

   Corrective actions are sought and recommended which can reduce the probability of future occurrences, or reduce their consequences.

3. A system for prioritization of the safety importance of the events which are analyzed. Priorities are based on relative assessments of safety impacts, or potential but plausible impacts, of the events.

4. A systematic and continuing evaluation of plant operating practices determined by plant evaluation visits.

   Balanced and trained teams of people who are highly experienced in plant management, operation, maintenance and technical support have visited and evaluated every operating commercial nuclear power unit in the U.S., and have started a second cycle of such visits.

"Confirmatory Research" Versus "Problem-Solving" Research and Development

The R&D program of the NRC may be regarded as involving two main kinds of outputs: (1) those related to "assist in establishing regulations for existing and future facilities" and (2) specific problem-solving related to providing information or data which can help exploit an opportunity for practical improvements, or solve well defined problems in design or in operation.(1)

The objectives and results from "Confirmatory Research" reflect the perceived needs and goals of the regulatory staff. Comments on this aspect of the R&D plan are difficult since they involve value judgements on the goals or means of regulation. The comments herein apply principally to the aspects of the R&D program, which are "problem-solving" in the sense noted above, including those which bear on or which parallel selected problem-solving by industry.

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(1) NUREG 0885

At the time the NRC was separated from the former Atomic Energy Commission, the NRC Research program was directed by 1974 legislation to restrict itself to "confirmatory research." In the last two or three years, this restriction apparently has been softened somewhat by a more permissive interpretation of the original regulation. Because of this history, only a part of the NRC program is applied to "problem solving" in the sense that the goals and expected results are formulated in such a way as to find direct application to make plants work more safety, reliably or efficiently. The balance of effort is still predominantly "confirmatory research" which is aimed at further defining regulations.

This policy still controls most of the formulation of much of the current NRC R&D program. For example, the 1982 policy and planning guidance document (NUREG 0885) states "the purpose of the research program is to assist in establishing regulations for existing and future facilities" (emphasis supplied). The question on this policy, I believe, underlies many of the reviews and comments on NRC R&D programs in recent years. Within the constraints of the existing legislation the NRC staff has become sensitive to the limitations of this pattern. There is now less emphasis on highly improbable and hypothetical situations or events. There is increased recognition in the FY 1984 program that the "limiting case" approach "not only may lead to unwarranted conservatism, but may result in design requirements that inhibit the ability to cope with actual accidents or precursors or potential accidents." This is an important and valid perception. In our studies of actual plant operating events, we often need to evaluate operating strategies and procedures which make best use of the information available to the operator, versus procedures or remedies based on assumed or hypothetical "worst case" situations.

The staff plan still further recognizes this trend by increased emphasis on the following topics:(2)

- "Better understanding of system response and safety margin provided by plants in real-world accidents and transients.
- "Analysis and appreciation of needed operational capabilities, operator training requirements, and human factors.
- "Knowledge of fuel behavior during transients and severe accidents.
- "Knowledge of fission product pathways, deposition mechanisms, and ultimate destinations.
- "Better use of probabilistic risk analysis with better ability to differentiate relative safety significance of regulatory issues or prospective requirements.
- "Understanding of a sound basis for the design of mitigating systems.

The emphasis on all of these topics is warranted, I believe. The first two of the topics above relate to a group of activities at INPO, for example the following:

Human factors, operator training, man-machine interfaces

INPO is working as technical staff support to substantial industry committees or working groups on these topics. In this area, INPO's efforts are to help make the best application of technology which is largely known, based on recent years of

(2) Quoted from draft NUREG 0784 p 1-3 and 1-4.

research at EPRI and useful results and publications in related fields including the aircraft industry, the space program, and some military programs.

Our emphasis is to give priority to those applications of human engineering which have a direct bearing on reducing the known elements of small or medium size accidents, based on actual operating experience which we have analyzed. Still greater priority is given to preventing or successfully recovering from accidents with potential for large consequences.

We endorse the use of some aspects of the Probabilistic Risk Assessment technique. This is to do *relative* assessment of alternate options - for example alternate remedies or procedures. The relative assessment can also set the relative priority and timeliness targets for implementing remedies.

INPO uses this approach to prioritize the Significant Operating Experience Reports, which we provide to all U.S. nuclear utilities (and participating utilities overseas). These are the results of screening, prioritizing and analyzing those events which involve significant damage, or reduction in safety margins. Also included are events which were benign in the actual case, but which have plausible variations which could be more serious.

For low-probability, high-consequence accidents, the probabilistic technique provides estimates that somewhat over half of the total risk exposure arises from a specific kind of operator error which can arise in unexpected situations for which a specific event-based analysis has not been done. The type of error involved is called "cognitive" error. This means that the operator misjudges the situation for any one of several reasons. He may then execute a set of procedures correctly, but without covering the actual situation. Skill-based or manipulative errors, we believe, do contribute to some operating events, but are less relevant to serious accidents unless they are compounded by cognitive errors. Specific event-based analyses are impractical for very low probability circumstances in which several failures may be involved - because the number of possible combination and permutations of such circumstances is impractically large. Accordingly, an approach which uses safe, symptom-based responses (Emergency Operating Procedures) has been formulated which does not depend on analyzing the exact sequences. This approach is endorsed in the NRC Action Plan (NUREG 0660) and is the subject of an intensive effort by all four of the industry owners groups also with staff support from INPO.

A principal human factors "finding" from the TMI accident, was that the distribution of critical information over a large area of the control room was diluted and confused by the multitude of alarms and the large volume of less important information (less important under disturbance conditions, but necessary for regular operation). The concentration of the most critical plant information into a compact set is now widely agreed to be responsive to this perceived need. This step is also responsive to the major "human engineering deficiency" being found in control rooms which can affect cognitive errors.

There is a complementary triad of developments which can provide a major gain in the reliability of operator response in managing rare but potentially serious events. These are:

(1) The symptom-based emergency operating procedures

(2) The computer-aided safety displays of key information,

(3) The requisite training to implement and maintain the continued capability to make effective use of emergency procedures and the key information.

The industry has come to the perception that the difficulties which have been experienced in the last three years in specifying this triad of activities, and which have delayed significant application of this major risk reduction option - can be surmounted. The essential perception is that no one of these activities is useful by itself. The triad of procedures, displays and training must be developed and proven out as a matched set. Each is complementary to the other and cannot be fully specified except with the other two.

Substantial efforts are now underway in industry, with INPO support, on the formatting and definition of Emergency Operating Procedures. Initially, these will be based on the normal control room, focusing operator's attention, by training and procedure, on the key information available. Later, this will be supplemented by a matching set of computer-aided displays (for example, Safety Parameter Display Systems, or their equivalents) and the appropriate modified training modules. The safety displays also lend themselves to effective computer-aided training models and techniques. This approach is now in concert with the NRC staff positions delineated in a recent staff paper.(3) The R&D plan in draft NUREG 0784 may benefit by further revision to take account of these developments.

Translation of R&D Results to Application

It is useful to review the relationship between research, development, application engineering, and actual use in operation. It is a common observation that the results of laborious and costly research and development often find very little actual application. This concern often has been raised by NRC research staff, and to a considerable extent is supported by the review of the NSCC Subcommittee. However, this situation is not the fault of research (nor of regulation). Even the most

(3) Secy 82-111, March 1982

successful results of research go to application only after a number of additional ingredients are provided. Once stated, these seem commonplace, but are nevertheless crucial. The additional essential ingredients include the following:

1. A perceived need (either problem to be solved or opportunity for improvement) by those who can apply the results in practice.

2. A priority for responding to the perceived need, relative to other pressing needs or requirements.

3. The resources to permit implementation. This often depends on the perception that the ratio of benefit gained to the resources required is attractive, relative to other uses of the resources.

4. The management resolve and capability to marshall the material and technical resources required to achieve implementation.

5. A regulatory climate of policy and practice which recognizes the need for practical trade-offs at each stage of formulation, planning, scheduling and implementation, based on factual assessment of real-world problems.

The work of industry working groups, EPRI, and INPO bear on the first four of these added elements. The FY 1984 R&D program for the NRC program shows many laudable steps in the direction of addressing real-world problems, the needed basis for Step 5 above. The lessons learned from INPO's continuing intensive review of operating experience, together with the findings and implications from the plant evaluation process can help provide factual inputs needed in planning and formulating R&D tasks which have increased likelihood of reaching actual application. INPO has a renewed memorandum of agreement with the NRC which provides for the exchange of basic information and analyses relevant to many of these topics.

It has been a pleasure to have the opportunity to speak to this subcommittee on this topic. There are a large number of other topics which are of direct interest to INPO and its related industry working groups; for example accreditation of training programs, criteria and evaluation of operations and of management, emergency response planning, radiation control and health physics. I have chosen to highlight only a few topics which seem to have large potential for gains in actual and in perceived levels of safety, and for which industry and NRC efforts are increasingly synergistic. We share with the members of the subcommittee and with the NRC, the common motives of helping to maintain a major element of the energy supply of the United States and to help to make it increasingly safe and effective in serving the public.

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B.S., University of Chicago, 1941
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Employment:

1944-47	Manhattan Project, University of California, Berkeley U-235 project; plutonium project
1947-52	Knolls Atomic Power Laboratory
1952-54	Knolls Lab; Project Engineer Submarine Advance Reactor PWR power plant for Triton Submarine
1954-57	Stanford Research Institute Manager, Nuclear Engineering Section
1957-63	General Electric Company, Advance Engineering, consultant and Director of engineering study group
1963-70	General Electric Company, Sunnyvale, CA. Manager of Development Engineering
1970-74	General Electric Company, Nuclear Division Manager of Nuclear Fuels Programs
1974-79	Electric Power Research Institute Director of Systems and Materials Department
1976-1977	Visiting Professor, Nuclear Engineering, Purdue University (Part-time)
March-April 1979	Deputy Director, Industry Advisory Group at TMI-2 site
1979-81	Electric Power Research Institute, director of Nuclear Safety Analysis Center
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Affiliations:

Sigma Xi
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Member, National Academy of Engineering (1981)
Member, Research Subcommittee of Nuclear Safety Oversight Committee (NSOC) Governor Babbitt, Chairman (1981)

Society Memberships:

Fellow, American Nuclear Society
American Physical Society
Member, American Association for the Advancement of Science
Member, Metal Properties Council
Member, Public Policy Committee ANS
Past Chairman, Planning Committee ANS
Past Member of Board and Executive Committee (2 terms) ANS
Fellow, American Institute of Chemists

Mrs. **Bouquard.** Thank you very much, Dr. Zebroski. I must comment on the fine testimony that we have received yesterday and today. It has been very well prepared. We do appreciate it.

NRC is studying the feasibility of an engineering analyzer with the goal of providing faster than real time operator analysis of LWR accident situations. Does this work relate to your programs?

Dr. **Zebroski.** Yes, to some extent. The question is one of sequencing the safety panel displays on our working systems. All of them have in mind the capability in the future of implementing additional modules for decision aids for the operators.

The main difficulty we see here is to have a good discipline to prove out the value of additional modules which the computers can now provide. As such calculation aids come into operation, if that they survive tests of practical usefulness, they should be implemented.

I should further comment that there are a few of these decision aids—logic diagrams—which are already easy to implement which are faster than real time and which will probably be used widely.

Mrs. **Bouquard.** Would you comment on NRC work at LOFT to evaluate improved instrumentation and advanced display systems?

Dr. **Zebroski.** I think that was addressed in the NSOC subcommittee report. The work has good objectives but I think it has been largely eclipsed by work much further along in industry so I therefore question it. It may have some benchmark value but it certainly will not go to much application.

Mrs. **Bouquard.** Very little practical application.

In speaking of NRC's human engineering program, yesterday ACRS expressed their belief that too much emphasis is being placed on research to support improvements in control room design and it is more appropriate to expend research effort toward the reduction of risk and human factors that have already been identified and are significant contributors. Do you agree with this assessment?

Dr. **Zebroski.** I agree with that observation and that is consistent with what I said. I can expand a little by saying that the major deficiencies are obvious. The information is dilute and under conditions where 100 or more alarms light up the most useful information is buried in the less important. Therefore, the concentration of the essential information I think almost everyone agrees to.

There is also an area that is called ergonomics—sort of time and motion study optimization, perception optimization. Some of these are useful. Many of them are convenient. But too many changes at once bring with them a serious problem of untraining and retraining people to new configurations. Therefore, I believe that these should be implemented only in a gradual and evolutionary way. The big step is to help the cognitive error under severe conditions.

Mrs. BOUQUARD. Do you think there are any reactor safety problems that should not be addressed in a research program—safety problems?

Dr. ZEBROSKI. There are tons of research results lying around that have not been used yet, there are areas in which the obstacles to application of already known technology are formidable. It is clear that further research in those areas will also have very little application.

I think one of the suggestions I read in one of the earlier pieces of testimony had to do with research on the self-examination of how the NRC might make its own implementation processes and regulatory processes more effective. Studying how to get a higher degree of application would be important in this respect.

Mrs. BOUQUARD. Dr. Zebroski, do you think there are any lessons that we can learn from safety research that is being done overseas?

Dr. ZEBROSKI. Yes; I think that this has to do with the difference in the prescription—the legislative prescription—on confirmatory research. I think that the overseas work runs more than 10 to 1 in favor of problem-solving versus confirming regulations. It is aimed at making things work better in some definable way and in reaching application. Perhaps only one-tenth as much effort is trying to define the problems better, define the regulations better. The ratio in the United States is in the opposite direction because of a historical situation in 1974 where you had to define a split of functions between NRC and the then ERDA—the remainder of the AEC. That split made some sense then.

If I may suggest an interesting thought for this committee—which would be an inquiry into the validity of the confirmatory research as the general guideline, that is research to define regulation. Inquiry into the effects of that policy and whether it is still productive for fiscal year 1984 as it was when it was drafted in 1974 might be a very interesting topic for this committee.

My personal feeling is that it has been rather counterproductive for the last 5 years at least.

Mrs. BOUQUARD. Thank you very much for your testimony. We appreciate your being with us today and if there are no further comments the subcommittee stands adjourned.

Our subcommittee will meet again tomorrow morning at 9:30 a.m. in room 2325 to hear from the Department on the advanced isotope separation program but this will be a classified hearing.

Thank you very much.

[Whereupon the subcommittee stood adjourned at 4:30 p.m.]

OVERSIGHT

THE NUCLEAR REGULATORY COMMISSION RESEARCH PROGRAM

TUESDAY, JUNE 15, 1982

House of Representatives,
Committee on Science and Technology,
Subcommittee on Energy Research and Production,
Washington, D.C.

The subcommittee met, pursuant to recess, at 1:02 p.m., room 2318, Rayburn House Office Building, Hon. Marilyn L. Bouquard (chairperson of the subcommittee) presiding.

Present: Representatives Bouquard and Lujan.

Staff present: John Dugan, staff director; Raymond J. Pennoti, professional staff member; Louis Ventre, counsel; and Joyce Gross Freiwald, professional staff member.

Mrs. Bouquard. The subcommittee will come to order. I ask unanimous consent that the use of recording devices be permitted during the proceedings of today's hearings.

Today, the subcommittee will hear from the Nuclear Regulatory Commission. It is my intention to focus these hearings on two themes: the availability of safe nuclear power to meet this Nation's economic and national security goals, and the most efficient possible use of American tax dollars.

I think that this subcommittee has already made impressive progress toward the enactment of a comprehensive nuclear waste management bill. We have worked to restore rational balance to the energy programs of this Nation, a balance between both the technologies such as solar, conservation, fossil, and nuclear, and between the impetus given these alternatives in the Federal budget.

While there are clearly unanswered public questions about key elements of the nuclear fuel cycle, I believe, as do a majority of Americans, that nuclear energy plays a significant role in the energy future of the United States and of other industrialized nations.

The availability of energy is intimately linked to economic prosperity, and to jobs. It is vital that the Federal Government foster such growth and economic expansion, and not hinder its development.

One of the most significant aspects of such a mutually beneficial relationship is the issue of the interface which exists between the private industry charged with meeting the industry demands of today and tomorrow, and the regulatory institution, charged with the responsibility of protecting the public health and safety. I, for one, do not accept that these responsibilities are either contradictory, or exclusive.

This subcommittee recently held oversight hearings on the NRC research program, and I directed the staff to provide the commissioners with my preliminary findings and recommendations from these hearings.

The programs examined by the subcommittee staff were generally of a high technical caliber. On the other hand, we are very concerned about the process used at the NRC to establish research priorities and to set program objectives. Neither annual nor long-term research plans appear to be developed within a sufficiently structured framework to insure that the considerable NRC research budget is used effectively. Program objectives are broadly stated in a way that does not adequately relate them to the regulatory process. Although the interface between the NRC and industry research programs is steadily improving, we feel that it could be strengthened considerably and that industry could and should be held accountable for increased reactor safety research. The relationship between NRC research activities and programs of the Department of Energy should be clarified to optimize use of Federal resources. Finally, research at NRC directed by offices other than the Office of Nuclear Regulatory Research should be better coordinated with the research office.

Despite these concerns, we have found that the NRC has recently been making steady progress in many of these areas. Some of our findings are recognized within the commission and remedies are being sought. It is with this understanding that we offer recommendations to the commissioners for their consideration. I intend to have the final version of the findings and recommendations documented in our subcommittee report.

We do hope that the subcommittee's hearings will contribute to our common goals: the availability of safe nuclear power for the American public and the most efficient possible use of our American tax dollars.

I do want to take this opportunity to thank Chairman Palladino, Commissioners Ahearne, Gilinsky, Roberts, and Asselstine for their willingness to meet with this subcommittee today.

Gentlemen, you may proceed with your testimony. We thank you for being with us today.

Mr. Chairman, you may proceed with your testimony as you wish, or you may summarize, and your testimony will be made a part of the official record.

STATEMENT OF HON. NUNZIO J. PALLADINO, JR., CHAIRMAN OF THE U.S. NUCLEAR REGULATORY COMMISSION, ACCOMPANIED BY HON. JOHN F. AHEARNE, COMMISSIONER; HON. JAMES K. ASSELSTINE, COMMISSIONER; HON. VICTOR GILINSKY, COMMISSIONER; HON. THOMAS M. ROBERTS, COMMISSIONER, U.S. NUCLEAR REGULATORY COMMISSION; DENWOOD ROSS, AND FRANK ARSENAULT, OFFICE OF NUCLEAR REGULATORY RESEARCH

Mr. PALLADINO. Good afternoon, Madam Chairlady and members of the subcommittee. At your invitation, my fellow commissioners and I are here this afternoon to respond to the findings and recommendations resulting from your recent review of our research program.

We generally agree with your findings. We were particularly pleased to read in your general findings a comment on the high technical quality of the work funded and directed by the NRC. This is most gratifying. It is obviously a major consideration in our program planning to assure that technical quality of the program remains high. We view our research program as an essential part of our regulatory efforts in that it provides a sound independent basis for judgment.

In recent years, the Commission has taken steps to redirect its research program. This redirection was prompted by a number of factors including the lessons learned from Three Mile Island [TMI]; the need to conduct our operations in a more business-like manner; a shift in perspective as to the relative roles properly played by NRC, DOE, industry, and foreign research; and a desire to improve the regulatory relevance of our research program. Your comments reflect an understanding of the need for this redirection and we appreciate your specific suggestions as to how we can move further and faster to achieve our goals.

This brings me to your first specific finding:

> Results of research directed by the office of nuclear regulatory research are not adequately reflected in the rules and regulations of the NRC.

We are making improvements in this area. The staff is drafting a revision to appendix K based on NRC-sponsored research to date which would provide some simplification of our requirements. One of the primary purposes for our research has been to provide the means for a realistic appraisal of the loss-of-coolant accident which eventually would be used for a second, more extensive, revision to appendix K.

Attached to our prepared statement is a description of some other examples of the ways in which we are using research results in the development of our regulations.

You also recommended that:

> The Office of Nuclear Regulatory Research should list regulations likely to be substantively impacted or confirmed by current and planned research programs between fiscal year 1984 and fiscal year 1988, the years covered by the long-range research plan.

The staff will incorporate this comment as they formulate the next long-range research plan. In addition, we publish in the Federal Register twice each year our regulatory agenda identifying our plans for revising our regulations.

Your second finding was that:

> The distinction historically drawn between "confirmatory" and "safety" research at NRC is artificial and counterproductive.

While in the past confirmatory research may have been overly emphasized, we feel that some confirmatory research is still necessary. For example, our programs in equipment qualification and fire protection are confirmatory in nature. However, we believe there is now a better appreciation for the full range of research tasks the agency should undertake. Within the current charter, the Office of Nuclear Regulatory Research conducts whatever research is necessary to support the NRC mission. However, to avoid potential conflicts of interest, the NRC does not develop specific safety systems that could subsequently be subject to NRC licensing.

Your third finding was that:

> The setting of research priorities at the NRC is an ad hoc process with no overall framework or systems overview and very little delineation of potential costs and benefits for individual programs.

We are continually working to improve our system of setting priorities and providing integrated research plans. For example, our severe accident plan is consistent with the intent of your first recommendation. Some remnants of an ad hoc process remain, however; and we agree that further improvement is necessary.

We believe that the benefits of overall program plans and the development or risk-based prioritization of safety research will be useful to all involved.

While we agree that our system of setting research priorities can be improved, we are troubled somewhat by the characterization of our current process as "dangerously fragmented between the user offices and the Office of Nuclear Regulatory Research." As a practical matter, we believe responsibility for setting research priorities must be shared by the user offices and the office responsible for conducting the research.

Your fourth finding was that:

> NRC's dominant role in U.S. safety research tends to direct licensee research resources toward assuring compliance rather than improving plant safety.

There is a cause-and-effect relationship between our regulations and industry research, but we do not see a strong cause-and-effect relationship between our research program and the industry research programs. We believe that, since the TMI accident, plant owners now recognize their strong financial interest as well as safety interest in preventing or mitigating accidents, and will tend to fund research in those areas. We fully recognize the need for the industrial community to play a strong role in safety research. We believe it is appropriate for licensees to expend resources, in part, in complying with our regulations. We expect most industry research will continue to be oriented toward improvements in reliability and other economically oriented issues.

The fifth finding stated that:

> Despite a strong recent trend toward increasing cooperation between the office of nuclear regulatory research and the industrial community, the current interface is not adequate for the NRC researchers to incorporate maintenance and operations considerations into their safety research.

Developing an understanding of the impact of the human element on safety requires input from organizations with actual operating experience. As a result, we continue to pursue cooperation with the Electric Power Research Institute [EPRI] and the Institute of Nuclear Power Operations [INPO].

For planning purposes, EPRI, INPO, and other research organizations were supplied copies of our long-range research plan last year and will be again this year. Their comments were invited and considered.

Your sixth finding stated that:

> The interface between the office of nuclear regulatory research and the Department of Energy is neither clearly defined nor well understood.

We support the need for an expanded role for DOE in safety research and agree with Mr. Minogue's May 18, 1982, testimony that DOE could and should compliment the NRC program. An agreement between DOE and NRC entitled "Interagency Programmatic Agreement in support of Improved Reactor Safety" was signed on December 28, 1979.

For both the management of nuclear waste and advanced reactor safety research—the Clinch River Breeder Reactor—NRC work is controlled to avoid unnecessary duplication of DOE work.

Your last finding stated that:

> Research activities at NRC, not directed by the office of nuclear regulatory research, are not adequately integrated with the overall research program.

Offices other than the research office do contract for work, but this is technical assistance in support of their licensing or regulatory functions. This is to supplement their technical capabilities on specific regulatory issues. For emphasis, we should state that for the most part, this technical assistance effort is used to solve short-term problems of immediate interest to the particular office, and it is not used on broad, long-term research programs.

Madam Chairlady, this concludes our prepared remarks. We will be glad to respond to any questions that the subcommittee may have.

Mrs. BOUQUARD. Thank you very much, Mr. Chairman. In your testimony, you referred to revisions to appendix K to reflect simplified rules and regulations. Would you describe the scope and the schedule of the near-term simplification of 10 CFR 50 appendix K that you mentioned in your testimony?

Mr. PALLADINO. It is my understanding that the staff has under preparation a proposal for a revision of appendix K that should be forthcoming, I believe, in a matter of 30 to 60 days.

Mrs. BOUQUARD. When do you expect the second revision to take place?

Mr. PALLADINO. That would require the completion of some of the research going on. I believe that is scheduled for about 2 years from now in 1984.

Mrs. BOUQUARD. Yes. Really, how much additional research do you feel is needed in this area?

Mr. PALLADINO. Well, perhaps, I ought to call on Mr. Ross to indicate what is going on. Then we can discuss how applicable it is.

Mrs. BOUQUARD. Mr. Ross, do you care to explain?

Mr. **Palladino**. Mr. Ross is the deputy director of our office of regulatory research.

Mr. Ross. Yes. I think that when Mr. Minogue and I were here last month we indicated that the loss of coolant-accident research could and should be—I think we used the word—"wound down" in the near future.

Mrs. **Bouquard**. That is correct.

Mr. Ross. Now, when we look at the 1985 timeframe—well, in the first place, the expenditures per year are monotonically decreasing. If we are successful in the 1984-85 timeframe in providing a substantive change in appendix K, then I think thereafter at most the research money that would be spent would be on the order of a few million per year, and it would be used to maintain codes in a proper working condition.

The computer codes used by the NRC staff to audit new applications—presuming, of course, that there are some—I don't see the expenditures going to zero. However, instead of $50 to $60 to $70 million a year, it may be 10 percent of that—on that order.

I believe that is responsive.

Mrs. **Bouquard**. Do you feel that the lack of funds is going to be detrimental to your overall goals?

Mr. Ross? I am sorry?

Mrs. **Bouquard**. Do you feel that your lack of funding or short fall in funding will be detrimental to your goals?

Mr. Ross. No. In the LOCA ECCS, we have no shortfall.

Mrs. **Bouquard**. All right.

Mr. Ross. We are spending in my opinion, I believe, an appropriate rate to reach the Commission goals of modernizing appendix K and on an appropriate time scale.

Mrs. **Bouquard**. Thank you very much.

We heard from Mr. Taylor at our previous hearings. He represented EPRI. He spoke of the need to predict the released magnitudes of radio nucleides to the atmosphere in the event of a degraded-core accident.

I think this is another area where our industry representatives have expressed an interest in seeing that we have research results applied to the regulations which in this case is a requirement for sitings and emergency procedures.

When do you perceive that these changes to the regulations will be proposed to reflect the new research results if any changes are proposed in this area, Mr. Chairman?

Mr. **Palladino**. Madam Chairman, we have been cognizant of the industry interest in this source term, as we call it. We have instituted research on the characteristics of deposition and form of various nucleides coming out as a result of an accident.

It is my belief that this is a program that, again, will take us until about 1984 before we will be prepared to make any changes in either our siting or our emergency plans.

Mrs. **Bouquard**. I was very impressed by your testimony that you had seven examples of potential future use of NRC research results in the development of regulations. In addition to the seismic designs of piping and severe accident, the list included safety equipment and fire protection and earthquake intensity and disposal of high-level waste.

When do you expect that regulations will be modified as a result of research in these areas?

Mr. PALLADINO. Well, I would expect that this would vary with each of the topics. Some of the results may not bring about changes in regulations because sometimes there is a desire to maintain some stability even if we find there is some additional conservatism.

Now, if you have a specific one in mind, we can try to address that.

Mrs. BOUQUARD. They will vary with the topics.

Mr. PALLADINO. Yes.

Mrs. BOUQUARD. And they would not necessarily necessitate changes in the rules and the regulations, but merely the putting of them into practice.

Mr. PALLADINO. Yes.

Mrs. BOUQUARD. You know, in the area of setting the research priorities at NRC, when we described it as being an ad hoc process with no overall framework or systems overview, I think that the NRC Long-Range Research Plan stated that it is intended to assist the Commission in establishing appropriate priorities and assuring effective utilization of our NRC resources.

Do you gentlemen here today find the current plan adequate for these stated purposes?

Mr. PALLADINO. Do we consider what adequate?

Mrs. BOUQUARD. That the long-range plan is adequate for the purpose in establishing priorities?

Mr. PALLADINO. Well, we are in the process of modifying the way we handle our research programs and planning. The Commission has issued its general requirements or general thrust in the policy and planning guidance that it prepared last year.

We also provide guidance to the staff as a result of discussion of specific issues such as the environmental qualification issue or the fire protection issue or even the TMI-2 accident.

Also, we do review the long-range research plan, and the ACRS in turn reviews it, and we benefit from their comments.

Occasionally, we also provide policy guidance through such activities as our severe accident review. We also have an opportunity to interact at the time of the budget. Contract approval is also another point at which we have another opportunity to react. Then, of course, the quarterly reviews on the specific issues is another opportunity.

Mrs. BOUQUARD. Commissioner Ahearne?

Mr. AHEARNE. The short answer is "No," and then I will give you a slightly longer explanation of that. No, because it is still a growing document. It was only a few years ago that the NRC attempted to put together a long-range research plan. We recognized the absence of that, the need for some kind of a planning document, some way to integrate what is essentially half of the agency's budget and most of the dollars that we spend.

It tends, as you know from having looked at it, to be spread over a wide number of programs. It is very difficult at the Commission level to try to see an overall framework for that.

We recognized that need and asked for the development of a long-range research program plan.

It has been a little difficult to get a hold of that partially because of the number of short-term efforts that research had to plunge in and participate with the rest of the agency as we tried to get many of the Three Mile issues resolved, but also because, over the last 3 or 4 years, we have had three directors of research. Each time a new director of research comes in, obviously that person takes a while to get their own priorities established and their own view before they are prepared to present it to the Commission.

The current plan is an improvement over the previous plan. I would expect at least another year's iteration before, at least for myself, I would say "Yes" to the question that you posed.

Mrs. **Bouquard**. My thought was that it would be difficult to evaluate priorities using that report.

Mr. **Ahearne**. That is correct. At the present time I would say that it is much more an attempt to pull together work that is there, as opposed to trying to put a framework down of priorities and then what pieces fit into those priorities.

However, I think the research people recognize that problem. That has been one of the issues that both we and the ACRS have raised to them.

As I tried to say, I think it is a big improvement over where they were, and I view it as sort of they are midway toward getting to the goal of putting together something that we can really use—we, and I guess others such as yourself—as a way of both making decisions on and seeing the whole context of the research program.

Mrs. **Bouquard**. Commissioner Asselstine, would you like to comment?

Mr. **Asselstine**. I don't really have much to add to what Commissioner Ahearne has just described. I would agree with the statements that he made.

I think that the long-range research program plan is a major step forward, but I think that the agency is in the refining process now to make sure that we do build into the plan not only our projections for what the research program is likely to be over those years but also to use that in comparison with the other programs of the agency to make sure that the research program does in fact fit in and support those other regulatory programs.

Mrs. **Bouquard**. Do you feel that it adequately explains why the planned programs are the correct ones?

Mr. **Asselstine**. I would agree with Commissioner Ahearne that right now I don't think that it does. However, I think that perhaps in the next iteration it will.

Mrs. **Bouquard**. I would like one additional comment on the cost and benefits of our research programs. Do you anticipate that this will be delineated in the future editions of the long-range research plans?

Mr. **Palladino**. Well, we really have no explicit cost-benefit criteria at the present time in our research program. There are a number of factors that have to be considered: For example, the risk significance of the work and along with that the uncertainties in the risk, the regulatory significance, and the cost effectiveness.

The criterion for doing research that relates to risk has to take into account the uncertainty. That it is useful for regulatory decisions should receive priority consideration in our work. Of course, the cost benefit—the cost and the yield factors—getting the most for our dollars—are equally important.

Actually, there is no neat equation for making these decisions on research. Risk significance, regulatory significance, and cost effectiveness are debated again and again throughout the agency and also in our discussions with the ACRS, the OMB, and the Congress, as well as with EPRI and other industrial representatives.

I would hope that as we improve our capabilities in prioritizing the work, I would hope that we can improve our cost-benefit relationship.

Mrs. **Bouquard**. Will these factors that you are using be made more explicit in reviewing the factors that you just mentioned?

Mr. **Palladino**. Can I make them more explicit?

Mrs. **Bouquard**. Can you make more explicit the factors that you are considering in your research programs——

Mr. **Palladino**. Yes. We could improve that.

Mrs. **Bouquard**. The factors in the research programs.

Mr. **Palladino**. Yes. I don't think that we have defined them as well as perhaps we have tried to define them in responding to your question.

Mrs. **Bouquard**. Thank you very much.

I notice that Mr. Lujan has joined us, and I would like to recognize him at this time for any opening remarks that he might have.

STATEMENT OF HON. MANUEL LUJAN, JR., A MEMBER OF CONGRESS FROM THE STATE OF NEW MEXICO

Mr. **Lujan**. Thank you, Madam Chairlady. I do have some prepared opening remarks, but I ask unanimous consent that they be included in the record.

Mrs. **Bouquard**. Without objection, they will be made a part of the record.

[Prepared remarks of Mr. Lujan follow:]

OPENING STATEMENT
THE HONORABLE MANUEL LUJAN, JR.
ERP HEARING ON NRC RESEARCH PROGRAM
JUNE 15, 1982

THANK YOU, MADAM CHAIRMAN. I JOIN THE OTHER MEMBERS OF THE SUBCOMMITTEE IN WELCOMING YOU, CHAIRMAN PALLADINO, AND THE OTHER DISTINGUISHED MEMBERS OF THE NUCLEAR REGULATORY COMMISSION.

AS RANKING MINORITY MEMBER OF THE INTERIOR COMMITTEE AS WELL, I HAVE BEEN PARTICIPATING IN THE CONFERENCE BETWEEN THE HOUSE AND THE SENATE ON THE NRC AUTHORIZATION BILL. THE TENTATIVE AGREEMENT FOR THE OFFICE OF NUCLEAR REGULATORY RESEARCH PORTION OF THE NRC AUTHORIZATION WAS FOR $240.3 MILLION FOR FISCAL YEAR 1982 AND $257.2 MILLION FOR FISCAL YEAR 1983. THESE NUMBERS INCLUDED $3.5 MILLION IN FY 82 AND $4.5 MILLION IN FY 83 FOR RESEARCH IN SUPPORT OF THE HIGH TEMPERATURE GAS COOLED REACTOR PROGRAM, AND $18 MILLION IN BOTH YEARS IN SUPPORT OF THE FAST BREEDER REACTOR PROGRAM.

THIS SUBCOMMITTEE HAS BEEN INTERESTED IN THE VARIOUS ASPECTS OF NUCLEAR SAFETY RESEARCH. I AM SURE WE ALL SHARE A CONCERN ABOUT GETTING THE MOST RESULTS FOR OUR RESEARCH MONEY AND THEREFORE WE WANT TO AVOID DUPLICATION IN THE RESEARCH PROGRAMS, WHETHER CONDUCTED BY NRC, DOE, EPRI, INPO, INDUSTRY, OR WHOEVER. I BELIEVE THAT PROGRESS HAS BEEN MADE IN THIS DIRECTION, BUT THERE STILL SEEMS TO BE ROOM FOR IMPROVEMENT. IT MAY BE THAT THIS SUBCOMMITTEE'S EFFORTS WILL PROVE TO BE A MAJOR STEP FORWARD.

However, in recognizing the areas where improvements are needed, we should never lose sight of the high quality work that is being done. I believe the Commission and this Subcommittee both can be pleased with the technical quality of the NRC research program. I feel that the use of the national laboratories by both DOE and NRC to conduct nuclear safety research is an important factor in both assuring high quality results and in avoiding some duplication. I also believe that the increased involvement of industry in nuclear safety research is to be encouraged.

Thank you, Madam Chairman. I am looking forward to hearing from our witnesses.

Mrs. **Bouquard**. Turning to another section of the list of our findings that were submitted to you, Mr. Chairman, NRC's dominant role in U.S. safety research tends to direct licensee research resources toward assuring compliance rather than improving plant safety.

My question to you is if licensees are required to conduct additional research to satisfy NRC's safety criteria, should the NRC research budget be reduced? Is this not duplicative?

Mr. **Palladino**. Well, we try to avoid duplication insofar as possible. I don't think any of our research ends up demanding more research by the licensees, but our regulations certainly sometimes do.

Generally, the kind of work that the licensees find more appropriate is to do those things that improve their economic position in the operability of the plant. However, sometimes these are very difficult to separate.

Mrs. **Bouquard**. Mr. Lujan, would you care to comment at this time?

Mr. **Lujan**. Yes, Madam Chairlady. Thank you. Thank you very much.

I did want to pursue a couple of points that I am not quite sure of. The difference between the research of NRC and DOE—is there a lot of duplication there or are the areas pretty clearly delineated? Whoever would care to answer it may respond.

Mr. **Palladino**. Well, probably not as well delineated as they ought to be. We do have a memorandum of understanding that I referred to in our opening remarks. They do set the general thrust of the directions that we want to take.

Actually, I think that the kind of research that DOE probably can make most contribution to is research that might add to the commercial feasibility of any actions that we in NRC are going to take.

It is appropriate for them, I believe, to maintain and operate large research facilities such as LOFT that could have broader applications to energy research than the narrow nuclear safety-oriented research objectives of NRC.

It could also assist in finding and developing solutions to safety issues, and most importantly in the development of new approaches to reactor systems.

The NRC's role is primarily that of being prepared to review whatever is proposed by the licensee and make sure that it meets the safety criteria that we have established.

Mr. LUJAN. Would NRC research be more in terms of confirming what is told you by the applicant, rather than looking into new areas that could be tried?

I am sorry, Mr. Gilinsky. You were starting to answer. Perhaps, you could bring it in together so that it all ties in.

Mr. GILINSKY. Well, I was going to say that in the past the country's light water research program was at the NRC. There wasn't any work to speak of at DOE on light water reactors.

Our work was restricted in our basic charter to what was called confirmatory research, and it was not contemplated that we would develop our own research laboratories. It was expected that we would contract for our work at what are now DOE laboratories, and then were AEC laboratories, and in the private sector.

Mr. LUJAN. Yes.

Mr. GILINSKY. What you have been talking about describes the conceptual difference in what might be conducted in one place and what might be conducted in another, but as a practical matter, it was all done at NRC.

Mr. LUJAN. Yes. Let me perhaps zero in on one area. In the 1982-83 NRC authorization which we are all trying to put together—as a matter of fact, we will have a conference at 2:30 p.m.—in the breeder reactor one of the concerns has been that in spite of all of the controversy, one of these days we will have some kind of a breeder reactor. Therefore, we put some money in there. I believe it was $18 million for that kind of research.

All right. We have the research going, as we know, on Clinch River through DOE. In what area or how would that $18 million be used by NRC as opposed to what DOE is doing now? That perhaps brings the question more in focus.

Mr. PALLADINO. I wonder if the amount is $18 million, or is it closer to $10.5 million? I think that the $18 million may include more than that. Am I correct?

Mr. ROSS. I could give some examples of where we could use that money and have used it——

Mr. LUJAN. Yes.

Mr. ROSS [continuing]. In what I call the audit role where the agency audits research by DOE. DOE conjectures an accident whereby the sodium would spill on concrete, and it reacts with the concrete and produces gasses and all sorts of bad side effects.

We did an audit role at Sandia Laboratories, also pouring sodium on concrete, and we got different results from DOE.

As a result, the two research organizations and the two sponsoring organizations have to get together and reconcile the difference. This is a classic case where the audit turned up something.

Mr. LUJAN. Yes.

Mr. ROSS. Now, there is another case where we have developed an extreme accident analysis code being used at Los Alamos to analyze disruptive accidents. We use this to postulate or audit the maximum energy that is used on the containment design basis which is one of our classical roles as a regulator.

Mr. LUJAN. Yes.

Mr. ROSS. Also, there is this fire protection research. Again, we view most of our work as an auditor, confirming what the licensee—in this case, DOE—brings to us.

Mr. LUJAN. Yes.

Mr. ROSS. That is, by and large, what we are doing.

Mr. LUJAN. Yes; I think that pretty well tells me what I wanted to know.

Mr. PALLADINO. May I make a clarifying point? I think that Commissioner Gilinsky earlier indicated that the Energy Reorganization Act limited us to confirmatory research and that was true; however, there was a change made in 1977 that broadened our capability to look at research that would help us improve the safety of the plants. Therefore, we are not as limited as we used to be to confirmatory areas.

Mr. LUJAN. Have you moved more into the area of original research as opposed to confirmatory research?

Mr. PALLADINO. We are moving into that direction.

Mr. LUJAN. Is that desirable?

Mr. PALLADINO. I think so. There is some confirmatory work that we still have to do, but understanding the phenomena with which we are working is very important, and using our understanding to develop codes to predict the performance of a system or component during an accident is very important. That is where I think we get our big payoff.

Mr. LUJAN. Yes.

Mr. GILINSKY. Mr. Lujan, if I may add a point, I think that it is useful for us to be able to develop an idea on a related research question up to a point.

Mr. LUJAN. Yes.

Mr. GILINSKY. If it then gets into a very large project, involving large facilities and great numbers of people, then it becomes something that is probably ripe for development by DOE.

Mr. LUJAN. By DOE?

Mr. PALLADINO. We do stop short of developing any system which we later would have to review. We don't believe that is our function. It may be an appropriate function for DOE and the industry, certainly.

Mr. LUJAN. We had a meeting here last week—I guess it was—with some Japanese legislators who were talking about some of the problems that they had and where they were going and where we were going. It was just in a discussion stage. One of the things that we were discussing was cooperative agreements of different kinds where they would pick up part of the tab also for particular kinds of research we have. I was informed specifically on the geothermal agreement that we have.

Does NRC have any cooperating agreements with other countries?

Mr. PALLADINO. Yes

Mr. LUJAN. In what areas?

Mr. PALLADINO. Well, we have a program with Germany called 2D3D.

Mr. Ross. And also Japan.

Mr. **Palladino**. And also Japan, I guess, is involved. Then we have one——

Mr. **Lujan**. What is 2D3D? What is the name of that little robot?

Mr. **Palladino**. Why don't I have Mr. Ross give you a summary of that?

Mr. **Lujan**. R2D2? Are they all related? [Laughter.]

Mr. **Palladino**. Mr. Congressman, I have the same problem with keeping up with these.

Mr. Ross. We have a large number of agreements around the world. In fact, Japan is one of the best places. We have in our staff resident engineers in some of Japan's research facilities in areas where they are building systems that are bigger and better than ours. They have staff engineers at our research facilities so that neither of us has to duplicate what the other is doing.

This is a bilateral agreement with Japan, and it is working very well. In some instances, it results in the inflow of money from Japan to us to help do the work. We must have at least a dozen agreements—some bilateral, some multilateral. We are developing more every month. We have people constantly abroad negotiating these agreements, most of which result to the financial advantage of the United States, I might add.

Mr. **Lujan**. Well, the reason I brought it up is, of course, the financial portion of it, but we all have the same needs, and for each one of us to invent the wheel is ridiculous when we could do it together.

Mr. **Palladino**. There is also an effort underway to extend work at LOFT by involving a consortium of interested countries, if it can be developed, whereby we can share the costs on future results coming from new LOFT tests.

Mr. **Lujan**. If we didn't have the budget constraints that we have now—putting those aside for the moment—what would NRC be more likely to engage in the area of research? What do you see as the crying need that we can't meet because of the budgetary constraints, or is there any? Are you totally happy with the budget?

Mr. **Palladino**. I am not sure that we have actually looked at our research program from that standpoint. Now, perhaps it has gone on in the past, but I might rely on some of the memory of my colleagues here.

Mr. **Ahearne**. I am sure, Mr. Lujan, you would get a different answer depending on which person you asked—whether it were the individual researcher in the research organization whose favorite program didn't quite make it through, and probably, Dr. Ross has got a different impression.

My sense, from 4 years on the Commission, is that we have never had something which we viewed as fundamentally significant that we have not been to get funded. Obviously, there are some programs that always drop out because either the staff was not able to convince the executive director that they were sufficiently well founded, the research office could not convince the Commission, the Commission could not convince OMB or the Congress. However, that is in the 10- or 15-percent range that are always a little soft, not really that well developed, not that well defended.

In an unconstrained budget, I imagine those would be there. Because of the basic character of research, probably there is some percentage of those—it is probably small; 1 or 2 percent maybe—that in the long term may have turned out to be well worth the funding.

I would expect that the programs that would fit into that category, if they don't make it one year, after more thought, hopefully, will be able to be better defended the next year and still make it.

Mr. LUJAN. Yes. I was thinking about HTGR, which was one of the items that we put more money into, that had no: made it in the request—things of that nature that you felt we might be short on—and the breeder that I mentioned a moment ago.

Mr. AHEARNE. Well, I am sure as you recall, Mr. Lujan, if you track back to where did the breeder drop out—sometimes it dropped out at the Commission level and sometimes it dropped out at the OMB level—to be consistent with an overall program that was being put in, in the long run, it ended up being there.

Mr. GILINSKY. Well, the HTGR, if I may add, Mr. Lujan, is certainly an interesting system with many interesting safety features. It comes down to a question of how much money do you want to spend to maintain that option, so to speak.

Mr. LUJAN. Well, we thought that $3.5 million for 1982 and $4.5 million for 1983 was kind of a good figure.

I have just one final question. Let me do some research of my own. In the operation of the Commission—and this is a little out of the scope of our hearing today—in most agencies there is the final decision by one person. Within the NRC, you don't have that. Some of us have felt that perhaps that tended to delay decisions.

Would the changing around from a commission to an administrator concept help in—all right, John?

Mr. AHEARNE. Yes. [Laughter.]

Mr. GILINSKY. Anyone who thinks so is free to resign.

Mr. PALLADINO. But there are days where it would be nice to be able to make a decision. That was one advantage of being a dean. You could listen to everybody and then you could say, "Now, this is the way we are going to do it."

However, I do think that we benefit from the thinking of five commissioners. Sometimes, I think we make better decisions as a result.

Mr. LUJAN. Well, it is kind of like the Congress. We worked on the budget last week, and I suspect there were 435 different budgets that each one of us would like to see enacted. As a result of that then, you don't end up with the real decision that you have to make. We ended up with a $99.25 billion deficit because that is what the leadership felt would pass. All of them were saying, "Well, we are not satisfied with that. We think we should have no deficit or $40, $50, or $60 or whatever the deficit may be."

The compromise then is a weakened position, perhaps, when you have to seek three votes rather than the final one making the decision.

Mr. Gilinsky, you started to rebut what Mr. Ahearne said?

Mr. GILINSKY. Well, you are asking whether one person would do it better. I suppose it depends on who that person is. There are a lot of people who would do it better.

Mr. LUJAN. In a dictatorship, it depends on who is the dictator.

Mr. GILINSKY. However, there are others who would not.

Mr. LUJAN. However, do you agree with the concept generally?

Mr. GILINSKY. I think that the commission form has served the country well. You have to strike a balance between administrative efficiency, which obviously is greater when you have one person, and sensible results which I think on the whole come out better when you have a commission.

Mr. LUJAN. Yes.

Mr. AHEARNE. This was a matter of considerable debate when the reorganization plan went through. There are, I think, a lot of arguments on both sides. I come down strongly on the side for our kind of a health and safety organization that a single administrator would be better.

Mr. LUJAN. Yes.

Mr. AHEARNE. That is not just in the research area, but across the board.

Mr. LUJAN. Thank you, Madam Chairlady.

Mrs. BOUQUARD. Mr. Roberts?

Mr. ROBERTS. I don't know whether you have got enough time. [Laughter.]

The chairman made an accurate point when he swore me in that, forgetting AEC, in NRC I am the only commissioner not to come from academia or from the Government. My total experience has been in the private sector, and this is a terribly inefficient organization in the commission form. I know there are arguments on either side, but I think it is a very inefficient process.

Mr. LUJAN. Yes.

Mr. ROBERTS. And I agree with Commissioner Ahearne.

Mr. LUJAN. Is there any other agency that—yes, I guess that the regulatory agencies generally have a——

Mr. GILINSKY. The Joint Chiefs of Staff.

Mr. LUJAN. What?

Mr. GILINSKY. The Joint Chiefs of Staff.

Mr. LUJAN. Yes.

Mrs. BOUQUARD. You know, we have many residents of the Third District of Tennessee who object to the TVA board only including three commissioners. They feel that the board should be expanded. Therefore, this is very interesting to me from that viewpoint because I represent the Third District of Tennessee where many of my constituents are asking for a board of five. Therefore, I am very interested from my own viewpoint to your reaction to the question.

I think that I am going to direct some of my questions again to the questions that were submitted to you because I feel that you should have your time in replying to the statements in a very eloquent manner.

We were talking about—and I think Mr. Lujan referred to it, too—the interface and things that should be done in balance but not in a duplicative manner.

In the work that you are doing in waste management, for example, to avoid duplication of efforts on the part of DOE, do you really think that the line is properly delineated? Do you think that it is documented—what your role is and the role of DOE?

Would you like to respond to that, Commissioner Ahearne?

Mr. **Ahearne**. Mr. Chairman?

Mr. **Palladino**. Well, there is an important point to be made with regard to this situation because, in this case, DOE is an applicant as well as being a development agency of the Government. We do want to hold DOE to the same terms in this case as we do to any applicant. Therefore, we are requiring them to develop their case, get the research necessary to support it, and bring it forth in a well-established and justifiable manner.

Mrs. **Bouquard**. The DOE is also responsible for the public health and safety. Would you not agree with that, Mr. Chairman?

Mr. **Palladino**. Well, DOE is trying to develop the product—the waste-management system and the packaging inplacement techniques. This is something that we would go to them for.

However, understanding the materials characteristics, both in the packaging, and understanding the geology of the area in which we are going to inplace this waste, understanding the ground flow of water or the hydrology of the site, understanding the stresses within the place where we are going to inplace the material—those are the kinds of things that we, as the NRC, need to help evaluate whatever DOE proposes.

Mrs. **Bouquard**. Well, the point that I was trying to make is, is there an agreement of what your roles are, and is this documented?

Mr. **Palladino**. I think there is pretty good agreement, and I believe we actually have a memorandum of understanding on that point as well.

Mrs. **Bouquard**. Mr. Ahearne?

Mr. **Ahearne**. Let me expand a little bit, if I can, Mrs. Bouquard. It is the interrelationship on the waste management between the NRC and the DOE which has been growing over the last 4 years because the DOE's program has really only been growing over the last 4 years. It shot up suddenly, as I recall, from maybe $40 or $50 million a year up to the hundreds of millions of dollars a year.

As they expanded very rapidly, the NRC found itself in a somewhat similar position. We had a relatively small group looking at waste management. We tried to similarly expand in people to get ready for this license application. Therefore, our people have tried to keep abreast of what DOE has been doing, and then, as the chairman pointed out, try to develop at least enough of a research program so our people would have a sufficient understanding of what DOE was doing and a sufficient understanding so that we could develop our regulations.

Unlike the light water reactor area, where there was over the last 10 or 15 years a body of regulations which were written, in the high-level waste management area, we are almost in the situation where the developer, DOE, is out doing research work developing a situation, and we are trying to write the regulations which they are going to have to meet at the same time. This probably has led us to do research work which in some ways is duplicative of what DOE is doing because DOE's time scale is to get ready to develop this system and NRC's was to get ready to write the regulations. Therefore, obviously, if you look at the two programs, there obviously were some places where there was an overlap of the same thing. However, it was because of these different purposes.

Mrs. **Bouquard**. Well, under the Atomic Energy Act—a point that I was also making—DOE is responsible for the protection of health and safety. Therefore, shouldn't they be treated just a little bit differently from the private sector that is mainly interested in profits?

Mr. **Ahearne**. Well, we certainly do treat them substantially different in the sense that there is a much closer working relationship between us and DOE. In fact, I can recall some Commission debates in which a proposal was made, which we did not accept, which was not to give them the kinds of benefits of doubts that we tend not to give to licensees. We do tend to presume that the DOE has a different fundamental purpose: to satisfy national objectives, that in some cases we do not presume a priori on the part of a licensee.

Nevertheless, there is going to come a point where they will come in with a license application. I think—I am speaking for myself and I believe it represents many of the people on the Commission and the staff—over the many years that the waste management program has tried to get put in place, the opposition mistrust on the part of States and local people with the Federal Government's approach has reflected itself in a lot of opposition that I don't think would go away if it were merely the Energy Department who were to come in and say, "We examined this system, and it is now ready to go in."

I believe there is some kind of a review process that is going to be essential to get an acceptance of whatever the DOE comes up with.

The way the law is now written, the NRC has to license that facility. We viewed our review process being that which we hope will get the public acceptance.

Mrs. **Bouquard**. Mr. Lujan, would you care to comment on that?

Mr. **Lujan**. Further on that, wouldn't the NRC role be that if DOE says, "We are going to put it in a particular geological media and it can stand this heat," and, you know, all of the things that they have found, wouldn't it just be the role of NRC to check that out and see as a matter of fact if what they are saying is correct but not to be doing research in all media. Although, on the other hand, I don't know how you prepare for it if you don't do research in all media.

Maybe what I am trying to say is after they make a presentation to you—to put it in this particular media—you would simply confirm whether their tests are correct or not?

Mr. **Ahearne**. Yes. I think that is basically correct. What we have been trying to do is to get enough of an understanding on our part so we would know, first, what kind of information would be necessary to make that kind of a demonstration, and, second, to be prepared to evaluate it. As you described it, that is basically the approach that we would take or I suspect we will take when that comes down.

Mr. **Gilinsky**. We also set the standards, Mr. Lujan, before they make a submittal, so that involves another kind of work.

Mr. **Lujan**. Yes, speaking of setting the standards, on mill tailing, you went ahead and set standards without really knowing what those standards should be. EPA was to evaluate the environmental concerns and then advise you, and NRC jumped the gun and required—at least in the minds of many and I happen to be one of those many—that the standards were excessive, simply so there wouldn't be any error. They were just set too high, and that way you would err on the side of too much regulation.

Mr. PALLADINO. Well, I wouldn't say that the NRC jumped the gun. We had a responsibility with which we were charged, and we were waiting and actually hoping and expecting that EPA standards would be set. However, lacking those standards, we couldn't go forward without setting our own.

Now, maybe they were too tight. I won't comment on that fact. However, we did have to go ahead, and we had to set some standards on which we could proceed.

Mr. LUJAN. Had you done extensive research on that? I remember no more than maybe a year and a half ago—you know, the previous chairman of the Commission said that it is more dangerous for you to smoke a cigarette than for you to stand on the pile for all day long. Then we were quite surprised by the amount of cover that NRC finally came up with as a standard.

Mr. PALLADINO. Well, I think that we still look forward to the EPA standards. Anything that could be done to expedite those, I think, would help the whole situation.

Mr. LUJAN. Is that kind of saying, "Well, if EPA comes out with different conclusions than we have reached, then perhaps, the changing of those orders would be in order"?

Mr. PALLADINO. Oh, yes. If they come out with different standards, we would have to incorporate them into our requirements.

Mr. LUJAN. Yes. How far are they on those? Do you know, Mr. Chairman?

Mr. PALLADINO. Well, I gather they have been in draft form for quite a while. Maybe you have a little more history on this, Mr. Ahearne.

Mr. AHEARNE. Mr. Lujan, in general with EPA standards, what we are always told when we deal with EPA is that they are almost ready to be published. That answer seems to persist over the years——

Mr. LUJAN. Yes.

Mr. AHEARNE [continuing]. Whether it is in mill tailings or high-level wastes—whatever standards. They are almost ready. I have long since tried to interpret that in terms of calendar days.

Mr. LUJAN. Yes; thank you, Madam Chairlady.

Mrs. BOUQUARD. Thank you very much, Mr. Lujan.

Mr. Ahearne, we referred to the need for a cooperative-review committee. Do you think that this would be valuable for us to make the best use of our tax dollars to improve the utilization of our research facilities?

Mr. AHEARNE. I am sorry. You will have to refresh my memory. A cooperative-review committee?

Mrs. BOUQUARD. I think that when you were answering a question that I asked you in regard to overlapping and duplicative efforts * * * that we felt a need for a cooperative-review committee. Is that correct—such as we proposed with DOE?

Well, let me put it another way.

Mr. AHEARNE. Do I endorse a cooperative review committee?

Mrs. BOUQUARD. That is the next question. [Laughter.]

Mr. AHEARNE. I think that having a closer working relationship with DOE would be very beneficial. Now, the current chairman will have to speak on the more recent experience, but I know that during the time when I was chairman and during the time that I can recall when Dr. Hendrie was chairman, we had difficulty getting a—I guess that the primary difficulty we had was trying to understand how DOE was going about its budget process. It never really seemed that we were getting a working relationship set up which would really enable a decision to be made on "here is a block of work that we both agree has got to be done." Now, who is going to do it? Is it going to be DOE? Is it going to be us? There was another player in that all the time, and that was OMB, who ended up having a significant say in where that work was going to be sent.

However, I think that some kind of a cooperative committee would be a good idea from my point of view.

Mr. PALLADINO. It is my understanding that such a group was formed as part of our memorandum of understanding that was signed in December of 1979. It called for a joint coordination group for improved reactor safety. As far as I know, it has been working, but perhaps Mr. Ross would like to comment on that.

Mrs. BOUQUARD. Mr. Ross, would you care to comment on that?

Mr. ROSS. Yes; the cited MOU does call for interagency meetings, and we do meet with DOE routinely. I believe, however, that the higher quality of interagency agreements comes from special agreements, and we have special working agreements being developed on the breeder research, on the LOFT consortium, and on the shared research associated with a realistic assessment of the radioactive source term. Therefore, we have general routine interagency agreements, and we have special agreements. Both seem to be working reasonably well.

At my level, we have frequent discussions with counterparts in DOE. and I think this is very helpful to both parties.

Mrs. BOUQUARD. Thank you very much. Let us turn to the area of research activities. One of the findings was that the research activities at NRC not directed by the office of nuclear regulatory research were not adequately integrated with the overall research program.

The NSOC reactor safety research review group stated that technical assistance programs are not usually defined as research even though some programs under the rubric of technical assistance have been long term and they have been exploratory in nature.

Would you comment on that statement, Mr. Chairman?

Mr. PALLADINO. Well, as I indicated in my prepared testimony, much of the work that goes on outside the office of research is technical assistance. When it becomes prolonged activity, it does assume some of the proportions of the research function, and I believe there would be benefit in looking at the whole picture when we look at research just to make sure that we haven't disguised research in some other form, or that we agree that should be in another form, but nevertheless be consistent with our overall objective.

Mrs. **Bouquard**. Well, I was interested in the distinction that NRC draws between the technical assistance and research because it seems to me that, given the lack of emphasis placed on deliverables and schedules by your research office, if I were a user office and I had a specific question regarding research, I think I would prefer to go the technical assistance route.

Mr. **Palladino**. Well, that isn't the way we go. For example, if one were trying to understand a phenomenon and develop predictive codes, I would say that is a research activity. However, if one needed to know the consequences of a certain action under a certain kind of accident, that I would say would be a technical assistance activity for which one could go to Oak Ridge or Sandia or somewhere else and get them to apply the developed codes to come up with a specific answer.

Mrs. **Bouquard**. Well, in regard to accidents, are you doing research in both places?

Mr. **Palladino**. Well, if one wants to know in a particular plant what would happen under a given set of conditions, that might be a technical assistance activity. If one generically wanted to know the phenomenon involved in a particular kind of flow pattern that develops during an accident and try to develop some predictive techniques, I would say that is a research activity.

Mrs. **Bouquard**. Well, if the Office of Nuclear Regulatory Research is responsive to the user offices, shouldn't it coordinate technical assistance programs as well?

Mr. **Palladino**. Well, we have not had it do so. I think there is a certain amount of importance in keeping the technical assistance in the hands of the office that has to have prompt answers. I think that the EDO has to maintain an oversight to make sure that these things don't expand into research programs. In our review function, we have that activity also.

Mrs. **Bouquard**. Well, it is my understanding that the office of nuclear materials safety and safeguards plans during fiscal year 1983 to develop technical information and to publish technical positions to provide guidance to DOE on the areas of waste form integrity and performance of shaft and bore hole seals and radio nucleide migration.

Now, is this work considered technical assistance, or is it considered research?

Mr. **Palladino**. Now, I believe—let me check. May I introduce Mr. Frank Arsenault who is the director of our division in the office of research?

Mr. **Arsenault**. Both research and technical assistance contracts lead to the publication of guidance for licensees. Where the object of the contract is to identify new—to look at new phenomena or gather new data, it is generally research project. Where it is the application of existing codes or the utilization of existing data, then it generally falls in the category of technical assistance.

Both types of contracts can be aimed at the publication of guidance to licensees.

Mr. **Palladino**. Frank, could you highlight how our waste management research is being coordinated?

Mr. **Arsenault**. Yes; I think there is probably closer coordination between the office of research and the licensing office in this case, the office of nuclear materials safety and safeguards, in the area of waste management than perhaps in any other area.

One of the reasons for this is that the organizational structure lends itself to that. Another is the proximity of the two offices since we are only a few floors apart in the same building. However, there is a constant—and I mean daily—interaction between the staffs of the two offices on the requirements for both technical assistance and research contracts. We formulate them and coordinate them continuously.

Mrs. BOUQUARD. Thank you very much. I would say that I really think that the difference between technical assistance and research is rather weak, and maybe more coordination is needed in this area.

I want to thank you distinguished gentlemen for sharing your time and your knowledge with us this afternoon. We would like to submit additional questions for the record, and we would ask that you reply so that they may be made a part of the record of these hearings.

Mr. PALLADINO. We will be pleased to do so, Madam Chairlady.

Mrs. BOUQUARD. The next scheduled subcommittee hearing is a joint hearing with the Subcommittee on Energy Development and Applications on national laboratories. These hearings will be held Thursday, July 15 at 9:30 in room 2318 of the Rayburn House Office Building.

If there is no further business, the subcommittee is adjourned.

[Whereupon, at 2:12 p.m., the subcommittee was adjourned, to reconvene at 9:30 a.m., Thursday, July 15, 1982, in room 2318, Rayburn House Office Building.]

APPENDIX I

STATEMENTS AND RELATED ARTICLE SUBMITTED FOR THE RECORD

May 6, 1982

Energy Research and Production Subcommittee
House Committee on Science and Technology
Room B-374
Rayburn House Office Building
Washington, DC 20555

Attention: Dr. R. J. Pennotti

Dear Dr. Pennotti:

In response to our telephone conversation I am addressing one aspect of safety-related research and development; namely, the application of conservative criteria and their impact on research and development. I shall limit my comments to piping systems because I am fairly conversant with them; however, by analogy the comments could be applied to other structures or systems. The following remarks are my personal opinions which have been expressed when on ACRS, to the NRC and to various portions of the nuclear industry. They do not necessarily represent the views of my employer.

- Piping failures, including the double-ended pipe break, have become embedded in the NRC criteria. This ignores the original application which was to size containments. Unfortunately, the DEPB became an article of faith, and the unresolved safety issue of asymmetric loads is a direct outgrowth. Since primary piping systems in essence are virtual pressure vessels, an instantaneous pipe break has a low probability of occurrence. I would place it at or below 10^{-6} reactor years. Unfortunately, DEPB assumptions bias R D programs.

- The effects of seismic loads on piping systems usually are modelled elastically because it simplifies computer analyses. Unfortunately, conservative damping factors lead to more and more snubbers being added. We have plants where the piping systems have been stiffened by the addition of so many supports that allowable tolerances on extended runs of piping are reduced to less than 0.1-in. radial which allows very little installation error. The incidence of supports or snubbers being pulled out of the wall during heat up or cool down confirm the existence of installation errors.

- An extension of the previous items pertains to snubbers added to dampen seismically induced vibrations. Hydraulic snubbers had poor performance records because of seal leakage, etc.; therefore, utilities went to mechanical snubbers. Some performed well; others tend to malfunction by locking up. Again, such lock up can lead to snubbers pulling out of the wall on heat up and cool down.

- Seismic experience in non-nuclear plants (e.g., petrochemical, fossil-utility, etc.) confirms that pipes will become inelastic and damp vibrations. Furthermore, there have been few if any seismically-induced failures in butt-welded pipe in a reasonable (e.g., not severely corroded) condition. This means that we may be using the wrong assumptions leading to somewhat irrelevant research.

- I probably am biased because of my close relationship to the SSMRP. Even so, I feel the program is well conceived because it attempts to model actual response of systems, not conservatively assumed values that bias ones understanding. I feel we will obtain a better understanding of failure mechanisms, separating the probable from the conservatively assumed. Furthermore, we should have a better estimation of the levels of seismic acceleration necessary to initiate a given failure mechanism. I have substantial reservations regarding some current Probabilistic Risk Assessment (PRA) programs because I am not convinced that the system is modelled correctly.

- An extension of programs aimed at piping response, primarily to seismic loads, is newly established under the Pressure Vessel Research Committee (PVRC). I chair the Steering Committee; we will attempt to obtain a better understanding of piping behavior in stiff and flexible systems, considering the implications of design and installation errors on day-to-day operation as well as the effects of extreme loads. There is a close cooperation between industry and the NRC at both policy and technical levels. unding is limited; however, there is expected to be a major voluntary effort. We hope to obtain a better understanding of piping response. If our initial assumptions are correct, it may be possible to modify existing reactor systems after appropriate documentation. We feel that more flexible systems (e.g., less supports, etc.) can be inspected and maintained better, should be more tolerant of design and installation errors, and we hope will exhibit minimal loss of system function during a seismic event. This is our hope. We expect to pursue the program objectively. If our assumptions are incorrect we will remodel our approach.

The preceding has rambled somewhat. Hopefully, it hasn't obscurred my basic tenet that research based on incorrect assumptions leads to a reinforcement of such assumptions because the experiments are designed to examine an assumed failure mechanism. While such research may be of good quality it is "bad" research in that it obscures actual response characteristics. I used piping as an example because of my familiarity. I believe the basic tenet applies to other systems as well. I have attempted to touch on seismic effects and good versus bad seismic research in similar fashion.

If you wish clarification or modification, please contact me. I expect to be in Washington May 20-21 and could discuss your areas of interest more fully.

Very truly yours,

Spencer H Bush

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June 14, 1982

Congresswoman Marilyn L. Bouquard
Chairman
Subcommittee on Energy Research
and Production
Committee on Science and Technology
U.S. House of Representatives
Suite 2321 Rayburn House Office Building
Washington, D.C. 20515

Dear Congresswoman Bouquard:

I wish to express my thanks for being permitted to testify at your May 18-19 oversight hearings on the Nuclear Regulatory Commission's research programs. At that time I spoke as a member of the Safety Research Review Group, that had been chaired by Dr. Norman Rasmussen. I think it may be useful to you to receive some further comments from me as an individual heavily involved in these programs, and in light of my background as the first director of the AEC's safety research programs, subsequently the first director of the NRC's programs.

These programs must be viewed in the light of the present state of the industry. The plants now being built have designs that were set more than a decade ago. The safety research program conducted by NRC is only nine years old, and its first solid results were obtained only about seven years ago. So there has not been any opportunity for the research programs to affect these designs.

Instead, the research programs have served to prove that the safety features of the plants (particularly the emergency core cooling systems) are conservatively designed. This achievement provides the basis for reassuring the NRC and the industry with respect to these aspects of design.

The further benefit of relaxation of requirements as the margin of conservatism is found to be substantial has not been realized. The reasons are very interesting. The industry has learned to live with the excessive conservatisms required by NRC, and would prefer not to start down any new paths that have unforeseeable consequences. Furthermore, plant designs are quite rigid, and it is difficult to see how economic benefit could be gained from relaxation of the conservatisms. For instance, plants are no longer built with stretch capacity in power level as they once were. Relaxing requirements imposed by NRC could perhaps lead to fuel management that would ensure longer fuel lifetime, but it is felt by many that efforts are better devoted elsewhere.

It is my personal opinion that the reassurance as to nuclear plant safety that has been provided by the safety research program has been of enormous value, but I also believe that the industrial views I have just outlined are too cautious. The ability now possible to alter practices in ways that would benefit the ratepaper should be explored seriously, and soon.

These are my general views as to programs that have been completed. I would also like to pass on some thoughts as to the present and future programs.

At present, almost all new NRC research programs are started as a result of having been requested by other parts of NRC. I do not believe there is a coherent structuring of programs to meet what can be foreseen as the long-term requirements of the NRC. A number of long-term questions can readily be discerned. These deal, for instance, with waste management, reprocessing, the use of plutonium in water reactors (which will surely return), the fast breeder economy, the effects of solution mining of uranium, decommissioning of plants, the possibility of recycling of reactor materials (e.g., steels), types of reactors that may replace water reactors in time, etc. This list is not exhaustive. Little is underway in any of these areas, which illustrates the point that the NRC programs do not have a forward-looking, long-term component that they once had. Unless this feature is restored, the NRC is committed to continuation of the policy of catch-up in its research programs.

I hope that these comments may be of some use to your Subcommittee on Energy Research and Production, in its role of oversight as to NRC research programs.

Sincerely,

Herbert Kouts

Herbert Kouts
Chairman

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The management of nuclear safety: A review of TMI after two years

by Thomas H. Pigford

The March 1979 accident at Three Mile Island (TMI) revealed that to have nuclear safety there not only must be reliable equipment, but also there must be competent and qualified people. At TMI there were equipment malfunctions, but the vital safety equipment performed well. An accident with radioactive release and serious core damage occurred because operators mistakenly stopped the flow of emergency cooling water. Health effects from radiation exposure were negligible. Continued loss of flow of emergency cooling water could have led to core melting, but the containment would have survived and protected the public. Serious fright and trauma resulted from technical errors and public announcements based on these errors a few days after the accident. Earlier experience from other reactors and analyses dating back to 1971 should have alerted the industry, the regulatory agency, and the operators and avoided the accident. This experience from the TMI accident provides many valuable lessons in the management of nuclear safety. Several such lessons revealed from the investigation by the President's Commission on the Accident at Three Mile Island are discussed here.

Lesson 1: *Plant equipment must be adequately maintained*

There are many examples of faulty maintenance that contributed to the accident at TMI.[1] A few are:

1. The TMI accident began with a normal event that occurs at any steam-electric plant: the loss of flow of boiler feedwater, followed by automatic shutdown of the steam turbine. **Auxiliary feedwater should have automatically flowed to the steam generators to cool the shut-down reactor, but the auxiliary feedwater valves were closed.**

2. Some of the most important events in the TMI accident center on the **reactor-coolant expansion tank** (see Fig. 1), **referred to frequently as the "pressurizer,"** because its water is heated to maintain the operating pressure of the reactor. For many weeks **prior** to the accident it was known that **there was a leak in the relief valve** on the expansion tank, the valve that opened during the accident to relieve reactor coolant pressure but failed to close. **Knowing of this earlier and uncorrected leak, the operators ignored symptoms** during the accident that should have led them to realize **that the relief valve was stuck open,** allowing reactor coolant to escape.

3. From post-accident sampling, it was found that charcoal filters to remove radioactive iodine from building air had deteriorated and were operating below design specification. Fortunately, the filters were still sufficiently effective during the accident.

Although these malfunctions were not the causes of the serious core damage and radioactive releases, they did distract the operators. Management of nuclear safety must include proper maintenance so that reliable equipment is always available and so that operators can diagnose important events when they occur.

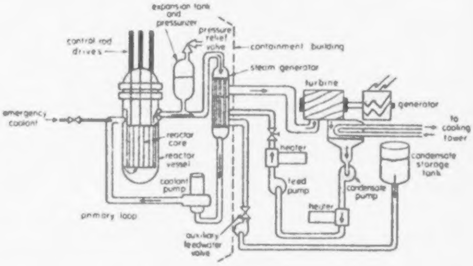

Figure 1 Schematic of the Three Mile Island reactor

Professor Pigford is a member of the faculty of the Department of Nuclear Engineering at the University of California at Berkeley. In 1979 he served as a member of the President's Commission on the Accident at Three Mile Island. This article is adapted from an invited speech at the Symposium of the Twentieth Anniversary of the Department of Nuclear Engineering, University of Tokyo, November 1980, and from speeches given at the ANS Topical Meeting on a Technical Assessment of Nuclear Power and Its Alternatives, Los Angeles, Calif., February 1980, and at the ANS/ENS Topical Meeting on Thermal Reactor Safety, Knoxville, Tenn., April 1980.

Lesson 2: *Reliable safety equipment is important, and it must be used properly*

When equipment malfunctions required the actuation of emergency cooling at TMI, the safety equipment performed promptly and reliably. The relief valve, shown in Fig. 1, opened automatically after the turbine stopped, and the reactor shut down a few seconds later. When the relief valve stayed open, some of the reactor coolant escaped into the containment, and the reactor pressure decreased. The emergency cooling water was injected automatically to cool the reactor. The malfunction of the open valve did not interfere directly with safety, because it provided a path for the heated water and steam from the emergency coolant to escape to the containment. The reactor was then being adequately cooled. Even the flow of auxiliary feedwater to the steam generators was not essential for reactor cooling and safety. If allowed to continue in this way, no core damage and radioactive release would have occurred.

Management of nuclear safety must depend, in part, upon reliable safety equipment. **People must not interfere improperly with the performance of safety equipment.**

Lesson 3: *Operators and engineers must be well trained with diagnostic ability*

The TMI operators were not properly trained. They and the supervisory engineers did not diagnose the elementary information of reactor coolant temperature and pressure, which was adequately displayed. They misinterpreted the trends of decreasing reactor pressure and rising water level in the expansion tank. This tank is also used to establish pressure for the reactor coolant. Not recognizing that the relief valve at the top of the expansion tank was open, and thinking that there was too much water in the reactor system because of the rising level in the expansion tank, the operators decreased the flow of emergency cooling water.

During the 108* minutes that it took for the water to boil away and to begin to uncover the reactor core, there was sufficient time for operators and engineers to diagnose the instrument readings and to learn that stopping the cooling water was slowly leading to a reactor accident with significant radioactive release. However, the operators had not been trained to recognize these conditions, even though such conditions had been predicted years before and had occurred at other reactors. Neither the operators nor the engineers recognized that the reactor coolant was boiling away. They were observing the reactor coolant pressure and temperature, but they did not understand the significance of this information. If they had simply injected enough emergency coolant to keep the water temperature below the boiling temperature, which is determined directly from the pressure, the accident would not have occurred.

Not much emergency cooling water was needed at that time. Although the emergency coolant was injected automatically at a flow rate of 3800 litres/min before the operators stopped the flow, a coolant flow as low as 600 l/min would have prevented core uncovery.

For adequate management of nuclear safety, operators must be properly trained. But training to respond to what we think might happen is not sufficient. There must be far better education, training, and understanding, and there must be more careful selection of qualified people. Operators and engineers must have sufficient diagnostic ability to deduce and respond to events that may not be the same as those prescribed in the training programs. They must have the understanding and capacity to diagnose information in terms of the functions of operating and safety systems, not just the identification of prepostulated events.

This lack of adequate diagnostic ability was also exhibited at TMI by the regulatory agency, as is discussed under Lesson 7. A new approach towards training and qualification of all people in the management of safety is required. At each level of responsibility there is some degree of diagnostic ability that is essential. The management of nuclear safety demands that this ability exist and that it be periodically and continually updated and verified. This is the most important lesson to be learned from the TMI accident.

Lesson 4: *There must be adequate procedures*

The known leak in the TMI relief valve caused the operators to ignore the written procedures and symptoms that should have told them that the valve was open. In normal operation some leaks could be tolerated if the procedures for an accident were properly written to take them into account.

There were written procedures at TMI to aid operators and engineers in their response to a loss-of-coolant accident, including a small-break accident such as a stuck-open relief valve. However, the procedures for the small-break accident were not adequate and could not be used. They mistakenly required symptoms that did not exist during the TMI accident, including loss of offsite power and loss of an emergency generator.

The TMI operators were misled by a legal specification that emphasized that the expansion tank should not be allowed to fill with water. They did not understand that this written requirement did not apply under transient and accident conditions.

These inadequate and incorrect TMI procedures contributed to the mistakes made by the operators and engineers. These procedures had been reviewed and approved by plant management and by the regulatory agency, but the deficiencies had not been recognized.

Management of nuclear safety requires careful attention to procedures, including their adequacy and the understanding of them by the operators and engineers.

Lesson 5: *The TMI safety equipment performed better than had been expected*

At 108 minutes after the operators reduced the flow of emergency cooling water at TMI the water had boiled away enough to begin to uncover the reactor core. Water continued to boil away and during the next 92 minutes,** until the operators restored the flow of emergency cooling water, there were times when all but the lower third of the fuel

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*The Nuclear Safety Analysis Center estimates 105 to 110 minutes for the beginning of core uncovery.

**NSAC estimates, after the beginning of core uncovery, an additional 90 to 95 minutes before emergency coolant was turned on.

TABLE I

RADIONUCLIDE INVENTORIES AND RELEASES AT THREE MILE ISLAND

Radionuclide	Core Inventory, Curies	FRACTION OF CORE INVENTORY RELEASED From Fuel[2]	To Building Air	To Atmosphere
Kr-85	9.6×10^4	0.47 to 0.70	0.6	~0.1
Xe-133	1.4×10^8	0.42 to 0.66	0.6	~0.1
I-131	6.5×10^7	0.41 to 0.55	7×10^{-5}	2×10^{-7}
Cs-137	8.4×10^5	0.45 to 0.60		0
Sr-90	7.7×10^5	$<8 \times 10^{-4}$		0
Ba-140	1.4×10^8	0.001 to 0.002		0
Assumed by NRC for reactor licensing[6]				
Kr-85 + Xe-133		1.0	1.0	
I-131		0.5	0.25	

was uncovered. The uncovered fuel became so hot from fission product heating that the zirconium cladding reacted with water vapor, which caused the fuel to heat even more. About half of the zirconium metal was fully oxidized. Enormous quantities of hydrogen gas were formed, most of which escaped into the containment during the next several hours and burned. The oxidized cladding and the oxide fuel probably crumbled into a rubble of cracked material, resting on top of the still intact lower portions of the fuel. None, or very little, of the uranium dioxide fuel is calculated to have reached its melting temperature of 2760 °C, although it is likely that small amounts of the fuel dissolved locally in the hot zirconium.

Millions of curies of radioactive noble gases, iodine, cesium, strontium, and other fission products were released from the fuel. The estimated[2] quantities of these fission products originally in the fuel and the fractions released from the fuel, to the air in the auxiliary and containment buildings and to the outside atmosphere, are shown in Table I. Although the radioactivity escaped into the air at a time when the meteorological conditions were poor for rapid dispersion, the resulting radiation doses to the surrounding population were negligible.

During the accident most of the radioactive noble gases were released from the fuel. The Xe-133 reaching the atmosphere was the main source of the small radiation dose to the public. In accident analysis, the radionuclide usually considered the most hazardous is radioiodine. Most of the radiodine released from the fuel remained in the coolant water, and only a relatively small fraction, less than 0.01 percent, escaped as a gas into the containment and auxiliary building.[1,3] The amount of iodine that escaped from the plant was between 10 and 20 curies, less than one millionth of that released from the fuel, and small enough to make only a negligible contribution to the radiation dose.

Earlier assumptions and studies by the former U.S. Atomic Energy Commission[4] suggested and assumed far greater core damage and greater releases of radioactivity from the fuel and into the containment air when a core is overheated under such degraded cooling conditions. The 1975 study by Rasmussen *et al*[5] assumed that core melting and loss of containment would occur in a small-break accident sequence involving operator error, similar to the initiating event at TMI. It predicted far greater releases from the fuel and to the outside atmosphere than occurred at TMI. The TMI release of iodine to the outside atmosphere was about a million times less.

For purposes of reactor licensing, the Nuclear Regulatory Commission (NRC) assumes[6,7] that an accident with highly degraded cooling conditions would release half of the inventory of radioiodine from the fuel and half of that to the containment air. The iodine released to building air at TMI was about 10 000 times less.

Equipment failures at TMI were not the real cause of the accident. Paradoxically, the accident was a demonstration that the safety equipment was effective and that it could perform better than had been expected. It demonstrated the validity of the multiple-barrier defense-in-depth approach towards reactor safety. The accident also indicated many areas wherein equipment modifications can result in further improvements in the safety of existing and future reactors.

Lesson 6: *The operating company and industry should be prepared to manage recovery from the accident*

Metropolitan Edison Company was seriously unprepared for an accident at TMI, as can be seen from the above examples. The company was not prepared to respond to the needs for informing the government and the public. The company had made no preparations for recovering from an accident, although it did quickly assemble an industrial team that demonstrated excellent knowledge and diagnostic ability. The industry was not prepared to handle the cleanup of radioactivity at TMI. Although basic technology was available at government laboratories, new process design and equipment are required to implement the cleanup from the reactor accident.

Management of nuclear safety must include better preparation for the recovery and cleanup from an accident.

Lesson 7: *A reactor accident can be serious even if radiation health effects are negligible*

Beginning the day after the accident, the NRC provided vital and needed communication with the public and with local government. The NRC made serious technical mistakes, however, that caused the accident to escalate into an incident with serious consequences. Two days after the accident the NRC misinterpreted information concerning a transfer of radioactive gas back to the containment. They made an incorrect calculation of the amount of radionuclides that would be released, drew an incorrect conclusion from plant surveillance data, and recommended evacuation. Pennsylvania's Gov. Richard Thornburgh decided not to accept the NRC advice, but public fears were escalated.

Two days after the accident the NRC learned from Metropolitan Edison that a large volume of hydrogen at 60 atmospheres partial pressure remained in the reactor vessel and coolant system. NRC Chairman Joseph Hendrie asked his staff about the possibility that oxygen could be generated by water radiolysis and could accumulate enough to cause a hydrogen explosion. The NRC staff did not know that in normal operation of a pressurized water reactor (PWR) hydrogen is added to the reactor coolant at a partial pressure of about 0.5 atm so that no oxygen will be formed by radiolysis, even under the intense radiation field of normal operation. The NRC staff was correctly informed both by the operating company and by the reactor designer that the hydrogen pressure of 60 atm was more than sufficient to suppress oxygen formation and that an explosive mixture could not form.

However, the NRC staff made its own calculations, which ignored the effect of dissolved hydrogen to prevent radiolysis. They accepted similar incorrect calculations from others, ignored data in their own files concerning oxygen generation in operating reactors, and mistakenly concluded that the hydrogen in the reactor vessel would soon become combustible. Three days after the accident the NRC announced this mistaken information to the public, along with concerns about possible rupture of the reactor vessel and containment and possible melting of the reactor core. Public fears were highly escalated, and there was no ready source of information to correct the errors made by the NRC. Shortly thereafter the NRC learned again from industrial groups that oxygen could not form and that the hydrogen bubble could not explode. The NRC informed the public of this correct information a few days later.

The President's Commission and its investigative staff concluded that the public fears, hysteria, and trauma, though short-lived, were the serious health effects from the accident. TMI tells us that a reactor accident that causes only negligible health effects from radiation can become serious because of public fright and hysteria.

The management of nuclear safety requires that the regulatory agency have sufficient technical knowledge and diagnostic ability with respect to the system that it is regulating. The regulatory staff should be able to identify and diagnose events important to public safety and should be able to obtain and interpret diagnostic information from reliable sources when needed. The NRC should recognize the need for this ability within its organization and should develop and maintain this necessary competence. The role of the NRC in the event of an accident needs to be defined.

Lesson 8: *Consequences of possible failures should be better understood*

If the TMI operators had not finally restored the cooling water, after the core had been uncovered for about one and a half hours, major fuel melting would have begun within another hour. The core was that close to melting during the day of the accident. The possibility of even later melting was raised by the NRC in connection with its hydrogen-bubble mistake three days later. There was no reliable information to guide the operating company, the NRC, and the public as to what to expect in the event of melting. Lack of such information encouraged much incorrect speculation and public fear.

Studies of the consequences of a melted core were terminated years ago. Even though the Rasmussen study had concluded that melting could occur under some accident conditions, it was evidently concluded by the AEC and the NRC that designing to protect against the large-pipe-break accident would preclude any reasonable chance of melting. However, TMI tells us that small malfunctions and human errors can compound into an accident perilously close to fuel melting.

Analyses carried out by the industry[8] and by the staff of the President's Commission[1] show that if the flow of cooling water had not been restored at TMI, a partially molten mass of uranium dioxide fuel and structural material could have destroyed the bottom of the reactor vessel and spilled onto the flooded containment floor beneath the reactor. Vessel failure and loss of all pressure in the reactor would have initiated the automatic flow of water from the other emergency cooling systems for the reactor and in the containment. The deluge of water contacting the hot semi-molten debris would have caused rapid overpressures of steam and even steam explosions, but the containment would have survived. The core-vessel debris on the containment floor would have been adequately cooled. It was concluded that even if this hot core-vessel debris were not water-cooled, the heat dissipation capacities of the 4-metre-thick concrete floor and the base pad would retain the semi-molten debris for at least several days, the core would eventually refreeze, and the containment would survive to prevent dangerous radioactive releases to the public.

The management of nuclear safety requires thorough factual analysis of events such as this, to provide a better understanding of consequences and for action planning in the event of emergencies. More and better factual information can displace much of the fear and speculation that interfere with the management of nuclear safety.

Lesson 9: *Accidents must be expected and planned for*

The TMI experience tells us that nuclear safety cannot be adequately managed on the basis of assumptions and beliefs that reactor accidents cannot and will not occur. Indeed, TMI confirmed the calculations in the 1975 study by Rasmussen,[8] who predicted that a small-break accident with operator error would occur by about the time of TMI, within the error bounds of the probability prediction.[3]

A lesson from TMI is that a safe low-risk system still has possibilities for accidents. Every technology imposes a degree of risk upon society, both in its routine operation and in the occurrence of accidents. Over a long enough time period even low-probability accidents will occur. We must be prepared for these accidents. The only approach that can result in public safety is to manage nuclear safety so that accidents have sufficiently low risk, reflecting sufficiently low probabilities or consequences.

Lesson 10: *We must learn from experience*

One of the most disturbing lessons from the investigation of the TMI accident is that the NRC and some parts of the industry had not given sufficient attention to earlier experiences that were direct warnings of the possibility of the TMI accident. Events similar to those that occurred during the first few minutes of the TMI accident, when the operators were becoming confused by the rising water level in the expansion tank and falling reactor pressure, had not only been twice predicted but also had twice occurred before the TMI accident.

In 1971 Westinghouse, which builds PWRs, but did not build the TMI reactor, predicted that a loss of reactor coolant from a break in the expansion tank could cause a confusing drop in reactor pressure without an accompanying decrease in water level in the expansion tank. Westinghouse notified the NRC of this possibility.[9,10]

The Westinghouse predictions were confirmed in 1974 at a PWR at Beznau, Switzerland, designed by the manufacturer. A relief valve in the expansion tank failed to close, the reactor-coolant pressure decreased, and the water level in the expansion tank increased and remained offscale. The Beznau operators diagnosed the situation early, however, and terminated the event before damage occurred. Westinghouse concluded that the reactor responded as expected from the analyses it had reported to the NRC in 1971 and 1972.

In 1977 Carlyle Michelson, an engineer then at the Tennessee Valley Authority (TVA), predicted that an open valve in the expansion tank could cause the rising water level and falling pressure in Babcock & Wilcox reactors, and he was concerned that this could confuse operators. His predictions were made known to Babcock & Wilcox, to the Advisory Committee on Reactor Safeguards (ACRS), and to the NRC. There was no followup by the NRC or Babcock & Wilcox. Although the ACRS did send, through the NRC, an inquiry concerning the Michelson prediction to a utility building a B&W reactor, the ACRS did not follow up after no response was received.

In 1977 a B&W reactor at the Davis-Besse plant in Ohio, almost identical to the TMI reactor, also experienced a turbine trip, a stuck-open relief valve, decreasing reactor pressure, and rising water level in the expansion tank. There, as at TMI, the operators mistakenly turned down the flow of emergency cooling water. Fortunately, after a short period the emergency cooling water was turned back on. The event was investigated and reported by representatives of the NRC and B&W. Neither organization alerted reactor operators so that a similar incident might be avoided at other nuclear plants.

If any one of these several earlier analyses and events, known to the NRC and to the TMI reactor supplier (B&W), had been recognized and followed up as indicators of a possible and likely accident, and if operators at TMI had been properly alerted, the TMI accident could have been prevented.[1] These are alarming examples of mismanagement of nuclear safety.

The management of nuclear safety should include careful and thorough diagnosis of operating events with the view that they may reveal precursors of new kinds of accidents that have not previously been analyzed or encountered. There must be a system to effectively disseminate this information to others who can aid in the diagnosis and who may be affected by the results. The past system in the United States is woefully inadequate. It can be corrected only if both industry and the NRC adopt the view that the understanding and dissemination of such information is vital to maintaining safety. It requires competent and qualified people, and it requires the full support of the industry and the NRC. Better dissemination of information will also require relaxation of the proprietary barriers of industry and a change from the adversarial attitude of the NRC.

Lesson 11: *Emergency planning should be realistic*

The TMI emergency plans requiring public action and evacuation were based upon unrealistically high radiation exposures specified by the regulatory agency for the large-pipe-break accident. There was little preparation within the community and state for responding to such emergency. They were poorly prepared to diagnose the public-health threat from the radioactive releases. From the TMI experience we learn that there must be contingency plans based upon different kinds of accidents, with different time scales and different radiation exposures.

Management of nuclear safety requires more comprehensive and realistic planning for emergency response, centralized authority for initiating and managing the offsite emergency response, and an adequate communications system for informing federal, state, and local agencies and for informing the public.

Lesson 12: *There should be adequate preparation for informing government and the public*

The TMI experience demonstrates that responding to the needs of the government and public for information during and after the accident can be a most difficult problem for the operating company. The government role of informing the public is equally difficult. The persons to provide this information must speak with knowledge and authority. Past approaches to providing public information are not adequate.

Industry and government must be better prepared to communicate to each other and to the public. News media must be competent to understand and report the information that they receive from these and other sources. They must be able to distinguish fact and informed opinion from speculation. TMI shows that careless and incorrect information, particularly when originating within the authority of industry or government, can exacerbate an accident of negligible radiation hazard to one of serious public concern.

Preparing for proper and adequate information transfer is a major problem in the management of nuclear safety. It may become one of the most difficult responsibilities of nuclear plant management and of federal, state, and local agencies.

Lesson 13: *Detailed regulations do not assure nuclear safety*

The TMI investigation revealed overreliance by the regulatory agency on an excessive number of detailed written regulations that seem to specify how a reactor should be designed and operated for safety. Writing these regulations and evaluating nuclear plants against these regulations has been the preoccupation of the regulatory agency. Developing the massive written responses required of each nuclear plant by these regulations has occupied much of the best talent within the nuclear industry. There have resulted the complacent and erroneous attitudes by the industry and by the NRC that safety results once these regulations are complied with. The investigation by the President's Commission concluded that this regulatory approach interferes with safety.

The Commission concluded that the most effective function of the regulator is to specify overall performance criteria for safety and to review and evaluate the success of the industry in achieving this necessary safety performance. This joint approach to safety, utilizing the complementary skills of regulators, designers, and operators, earmarked the early development of safety systems. It placed the greatest and ultimate responsibility for achieving safety on the designers and operators, where it must remain. This fundamental approach to safety has become dangerously submerged by the present regulatory practice.

There are many other problems with the current U.S. regulatory approach. The regulations themselves have not been developed with an adequate focus on what are the most important elements of safety and how they are to be achieved. These regulations and guidelines have been developed without quantified safety goals and objectives. Many safety concerns have been postulated and acted upon without adequate evaluation of their importance to safety or of their consequences. Present and past approaches to U.S. regulation do not exhibit a reasonable determination of priorities or allocation of resources in proportion to the estimated risk to the public. A disproportionate effort on the part of government and industry has been required for some issues that have only a marginal impact upon risk. Many of the requirements are arbitrary and are mandated without a valid technical basis or value-impact analysis.

Many of these problems reflect lack of determination by the regulatory agency and by government of the real purpose and goals of nuclear-safety management. Without any criteria or yardstick for "how safe is safe enough?" there are no clear means for the NRC to set priorities according to risk and degree of safety achieved. Lack of such criteria invites overly zealous demonstration of concern for safety by regulators, particularly under the scrutiny of public criticism, with little or no balancing of the safety benefits against the cost of achieving them. Too little attention has been given by the regulators to the costs of delays and postponements in resolving issues.

The actions and problems of the NRC reflect a staff and management with too little practical experience in designing and operating equipment similar to that which they regulate.

Another problem in the present approach to nuclear safety is its stifling adversary nature. What is mainly a process of technical specification, design, and review for safety has become submerged in the adversarial legal process of the NRC. The existing U.S. process inhibits the interchange of technical information between the NRC and industry. Lack of such interchange was one of the primary contributors to the TMI accident.

In addition to insufficent evaluation of operating experience, as has already been discussed, the present regulatory system suffers because it is not based upon a comprehensive approach to the entire plant. A large proportion of the regulatory staff are specialists focusing upon narrow topics. There are relatively few systems engineers with the experience and qualifications to integrate individual safety features into an overall concept and to place issues in perspective.

Many of the problems of diagnosing events relevant to reactor safety can be attributed to the overwhelming emphasis within the NRC on the use of conservative models and assumptions. Although it is important to ensure that expected operating conditions are sufficiently removed from failure limits, this can be achieved by means other than intentionally distorting the equations of physical phenomena, which must remain as tools to interpret and understand what is really happening. Safety is compromised when responsible professionals rely instead upon equations, models, and computer codes with buried and forgotten distortions, made nonrealistic for the purpose of incorporating design margins.

Management of nuclear safety requires a competent regulatory approach that is aimed at identifying the main goals and problems of safety, that specifies and evaluates safety performance, and that is implemented by realistic analysis and diagnosis.

Lesson 14: *The operating company is primarily responsible for nuclear safety*

Even the most competent of regulatory agencies cannot assure nuclear safety. Primary responsibility rests with the operating company, together with the engineering firms and equipment suppliers that design and build the plant. The operating company must expend sufficient manpower and energy to go beyond fulfilling the requirements of the regulatory agency. It and its suppliers must have qualified manpower to attend to all problems of nuclear safety, whether or not they are required by regulations. They must take the initiative in innovating ideas and procedures to maintain and enhance nuclear safety.

The performance history of the U.S. nuclear reactor industry

These are some of the principal lessons learned from the Three Mile Island accident that are important to the management of nuclear safety. These remarks are critical of many parts of the U.S. nuclear reactor industry and nuclear regulatory agency. However, these criticisms should not obscure the fact that in over 500 reactor-years of commercial nuclear power operation in the United States there has still been no identifiable effect upon the physical health of the public due to radiation and radioactivity from nuclear power plants. This outstanding record has been achieved by the industry and the Nuclear Regulatory Commission—the parties that have been criticized—and under the system that has been criticized. These critical remarks should be considered not as a condemnation of the parties and system, but rather as recommendations for improving the management of nuclear safety so that the enviable record can be maintained.

What has been learned by the industry?

The extent to which government and industry in the United States are solving the fundamental problems revealed by the TMI investigation cannot yet be fully evaluated, but it is timely to comment on action and lack thereof.

Among the many responses of the electric utility industry to TMI are its creation of the Nuclear Safety Analysis Center (NSAC), the Institute of Nuclear Power Operations (INPO), and the Nuclear Electric Insurance Limited (NEIL).

NSAC was created shortly after the accident to analyze the TMI events and evaluate the implications to nuclear safety. Technical analyses by NSAC, as well as analyses by individuals in the Department of Energy laboratories, were of considerable benefit to the President's Commission in carrying out its investigations. NSAC continues as a technical analysis center for reactor safety. It is evaluating the licensee event reports (LERs), which have been submitted to the NRC by operating utilities for years, but, prior to TMI, not subjected to systematic evaluation. NSAC carries out case studies of safety-related events, such as the loss of electrical power to nonnuclear instrumentation at Crystal River and the partial failure to scram at Browns Ferry. It is performing generic safety studies such as consequences of degraded cooling of a severely damaged core and core melting. NSAC operates an information clearinghouse, also used by the NRC.

The later creation of INPO was aimed at addressing one of the central problems revealed by the TMI accident, the need to have qualified and competent people to operate and manage nuclear power plants. INPO is to establish standards of quality and competence, and it is to regularly evaluate operating practices at nuclear plants. It is to set criteria for personnel qualification, training programs, and instructors. It will review and analyze operating experience, together with NSAC, and will review utility emergency preparedness and the human aspects of plant systems design. The INPO program is a massive and difficult undertaking of self-improvement by the electric utilities that operate reactors. The intensity with which the industry has initiated this venture indicates its intent that it be successful. It will not be easy, however, for INPO to influence changes in the management approach to safety, where needed, among so many different member companies.

TMI has become a dramatic demonstration that a major loss of nuclear generating capacity can be a crippling financial burden to the utility. Replacement energy must be purchased from more expensive sources, without corresponding increases in authorized revenues. After TMI the utilities established the mutual-insurance organization NEIL to protect against the cost of replacement energy. NEIL expects that standards set by INPO and results of INPO examinations will affect the insurance coverage offered by NEIL to its member companies. A requirement by NEIL that a utility meet INPO standards would give authority and teeth for INPO to carry out its mission.

The industry has also set up several teams to study and implement the many new technical and operating changes that have arisen from the TMI experience.

The industry has taken the initiative in remedying the specific technical and operational problems revealed by TMI. It is setting and policing its own standards of excellence to ensure the effective management and safe operation of nuclear power plants. However, industry is still being swept along by the NRC's continuing emphasis on nuclear safety by regulation, the approach found by the President's Commission to interfere with nuclear safety. Industry's concern with meeting the formalism of NRC regulations is still inhibiting and throttling new ideas and technical innovations more directly related to safety. Evidently, industry lacks either the will or the strength to lead, by itself, the more performance-oriented approach to nuclear safety recommended by the President's Commission.

What has been learned by the government?

In response to the recommendations of the President's Commission, President Carter proposed some reorganization of the NRC to give its chairman greater management authority. The President's proposals were far short of those recommended by his TMI Commission. After negative reactions from many in Congress and from most of the NRC commissioners, President Carter submitted a modified reorganization plan, which was adopted in the absence of disapproval by Congress. The modified plan, implemented in October 1980, gives the NRC chairman and the Office of Executive Director somewhat greater authority over NRC operations. However, there is little resemblance here to the changes recommended by the TMI Commission.

President Carter proposed no specific changes or NRC action to remedy the more fundamental problems of the NRC regulatory approach, such as those identified here in Lesson 13, but he directed the NRC to study these issues. There has since been little apparent response by government to the conclusions raised by the President's Commission and by other TMI investigations that fundamental changes are necessary in the NRC regulatory approach.

President Carter directed the NRC to proceed with licensing review. The NRC had reacted quickly to TMI by imposing a moratorium on its review of applications for operating licenses and construction permits, while it studied the accident events and specified detailed requirements to remedy the immediate faults and malfunctions revealed from the TMI accident. In June 1980 the NRC resumed licensing by authorizing low-power testing of the TVA's Sequoyah reactor, followed later in 1980 by authorizing the full-power operation of Sequoyah and a few other reactors. However, operating licenses for several other completed nuclear plants have been long deferred. Even for those plants for which essentially all post-TMI technical issues appear to be settled, such as Pacific Gas and Electric's Diablo Canyon plant, delays due to the NRC's internal procedures offer little promise of licensing until 1982.

President Carter created a Nuclear Safety Oversight Committee, along the general lines recommended by the TMI Commission. The committee has begun its review of the NRC and industry approaches to nuclear safety. It recognized the safety and licensing significance of the low-radioactivity releases exhibited at TMI, as discussed in Lesson 5 and shown in Table I, it inquired into similar data from other experiences on release of radioactivity from degraded cooling conditions, and it recommended to President Carter that this new information be seriously considered in licensing and in emergency planning.

ACRS actions since TMI indicate an appreciation of the problems of the present licensing approach of the NRC. The ACRS has suggested a form of a safety goal; it has urged use of probabilistic risk analyses and cost-benefit considerations to set priorities in reactor safety; and it has urged the use of performance criteria as a means of achieving and regulating safety. It is not yet clear, however, that the ACRS recognizes the extent of the faults and failures in the present regulatory approach, nor is it clear that ACRS can and will influence changes within the NRC.

Some reorganization within the NRC has occurred. One new group is to monitor and diagnose operating events; another is concerned with human factors, such as those arising in control room design. The NRC staff has been diligent in identifying many technical issues that arise from the TMI experience. Programs and evaluations are under way that seem to address all of the technical issues identified in the TMI investigations.

The NRC's approach, however, offers little promise of timely resolution of many of these issues. The need and means for coping with greater quantities of hydrogen from zirconium oxidation remain undecided. The performance of reactor cores under degraded cooling conditions is now to be subjected to a protracted public hearing and is several years away from resolution. The NRC stance can be reasonably viewed by the utility industry as an ever-widening opening of new issues and presently unknown future requirements for licensing. It is understandable that utilities are reluctant to commit toward additional nuclear construction until they receive clearer indications of criteria to be fulfilled in obtaining a construction permit or an operating license. If the NRC does find some effective and realistic means of factoring quantified safety benefits and costs into its approach to regulatory standards and licensing review, it should give greater recognition to the very large costs of delaying resolution of safety questions and regulatory requirements.

Even more disturbing is the evidence that the NRC shows little recognition of the fundamental flaws in its approach to reactor safety. The NRC has finally initiated a study that may eventually lead towards some formally established safety goal or standard. There is no evidence, however, that the NRC regulates by measuring performance against the less formalized safety goals that have been suggested or adopted in risk assessment studies. There is little evidence of NRC initiative to regulate by setting performance requirements for safety and by evaluating industry's designs and experience in meeting these performance requirements.

Instead we see continuing promulgation of detailed regulations by the NRC staff to mandate designs, design reviews, additional safety systems, and revised operating procedures that are assumed to result in increased safety. This approach continues to preoccupy the NRC with setting and determining compliance with detailed requirements, in the same way that it ties up too much of the available manpower within the industry in responding to these detailed requirements.

A new approach aimed more at achieving safety through performance specifications, with priorities toward safety achieved and measured against some realistic and achievable safety goal, would bring into question the need, relevance, and validity of much of the voluminous detail of NRC regulations and regulatory guides. It is understandable that the NRC staff may be reluctant to undertake a new approach that can undermine its extensive investment in regulation by detailed documentation. But the cost to the NRC of a responsive implementation of the strong recommendations by the President's Commission, as well as recommendations by other post-TMI investigations, would be far less than the cost to the country of continuing and intensifying the present and past regulatory approach.

The need to initiate fundamental changes in the regulatory approach to reactor safety is one of the clearest lessons from TMI, but it is apparently one of the most difficult lessons to understand, accept, and implement.

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APPENDIX II

ADDITIONAL QUESTIONS FOR THE RECORD

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 14, 1982

Dr. John V. Dugan, Jr.
Staff Director
Subcommittee on Energy Research
and Production
Committee on Science and Technology
United States House of Representatives
Washington, D.C. 20515

Dear Dr. Dugan:

In response to your letter of May 4, 1982 to Mr. Robert B. Minogue of the NRC staff, enclosed please find the NRC staff response to the questions posed in your letter.

Please note your additional question (unnumbered) on funds allocated to universities is included at the end of the enclosure.

If this office can be of further help, please do not hesitate to contact me.

Sincerely,

Carlton Kammerer, Director
Office of Congressional Affairs

Enclosure:
As stated

QUESTION 1. Please provide a cost breakdown of FY 83 research funding to individual program elements.

ANSWER

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

	FY 82	FY 83 CONG.
LOCA & TRANSIENT	$ 30.9	$ 30.0
LOFT	42.0	15.0
ACCIDENT EVALUATION & MITIGATION	33.1	47.2
ADVANCED REACTORS	7.5	13.0
REACTOR & FACILITY ENG.	33.8	38.0
FACILITY OPER. & SAFEGUARDS	13.0	13.5
WASTE MANAGEMENT	12.2	13.6
SITING & ENVIRONMENT	9.0	9.0
SYSTEMS & RELIABILITY ANALYSIS	15.1	15.9
TOTAL PROG. SUPPORT	$196.6	$195.2

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

LOCA & TRANSIENT	FY 82	FY 83 CONG.
SEMISCALE	$ 7.5	$ 8.1
SEPARATE EFFECTS EXP. & MODEL DEV.	6.0	5.8
3-D PROGRAM	6.0	6.5
CODE APPLICATION & IMPROVEMENT	8.9	8.7
FUEL BEHAVIOR UNDER OPER. TRANSIENTS	2.5	0.9
TOTAL PROG. SUPPORT	$30.9	$30.0

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

LOFT	FY 82	FY 83 CONG.
TEST OPER.-FAC. SUPPORT-SHUTDOWN	$30.5	$11.1
ENG. & ANALYSIS	8.3	2.0
FUEL PROC. & EXAM.	3.2	1.2
PROJECT CLOSE OUT	-	0.7
TOTAL PROG. SUPPORT	$42.0	$15.0

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

ACCIDENT EVALUATION & MITIGATION	FY 82	FY 83 CONG.
BEHAVIOR OF DAMAGED FUEL	$18.0	$24.9
FUEL MELT	7.1	10.3
FISSION PRODUCT RELEASE TRANSPORT	4.1	7.1
ACCIDENT MITIGATION	1.6	1.5
SASA	2.3	3.4
TOTAL PROG. SUPPORT	$33.1	$47.2

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

ADVANCED REACTORS	FY 82	FY 83 CONG.
FAST REACTORS	$ 5.0	$10.5
GAS-COOLED REACTORS	2.5	2.5
TOTAL PROG. SUPPORT	$ 7.5	$13.0

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

REACTOR & FACILITY ENGINEERING	FY 82	FY 83 CONG.
MECHANICAL/STRUCTURAL ENG.	$11.0	$12.5
PRIMARY SYSTEM INT.	15.3	17.4
ELECT. EQUIP. QUAL.	3.5	3.8
FUEL CYCLE FAC. SAFETY	1.3	1.2
EFFLUENT CONTROL & CHEM. SYS.	1.0	1.4
DECOMMISSIONING	1.7	1.7
TOTAL PROG. SUPPORT	$33.8	$38.0

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

FACILITY OPER & SAFEGUARDS	FY 82	FY 83 CONG.
HUMAN ENG./MAN-MACHINE	$ 4.2	$ 5.1
PLANT INST. & CONTROL	2.7	3.6
OCCUPATIONAL PROTECTION	2.0	2.1
EMERGENCY PREP.	0.5	0.5
SAFEGUARDS	3.6	2.2
TOTAL PROG. SUPPORT	$13.0	$13.5

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

WASTE MANAGEMENT	FY 82	FY 83 CONG.
HIGH LEVEL WASTE	$ 5.6	$ 6.1
LOW LEVEL WASTE	4.0	4.3
URAN. RECOVERY	2.6	3.2
TOTAL PROG. SUPPORT	$12.2	$13.6

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

SITING & ENVIRONMENT	FY 82	FY 83 CONG.
EARTH SCIENCES	$ 5.1	$ 5.5
SITING & ENVIRONMENT	1.2	1.5
HEALTH EFFECTS	1.5	2.0
ENVIRONMENTAL IMPACTS	1.2	0
TOTAL PROG. SUPPORT	$ 9.0	$ 9.0

NUCLEAR REGULATORY RESEARCH
(DOLLARS IN MILLIONS)

SYSTEMS & RELIABILITY ANALYSIS	FY 82	FY 83 CONG.
RISK METHODS & DATA EVAL.	$ 7.0	$ 7.0
REACTOR RISK & REL. ANAL.	6.6	7.4
TRANSPORTATION & MATL. RISK	1.5	1.5
TOTAL PROG. SUPPORT	$15.1	$15.9

QUESTION 2. Please describe the PWR instrumentation being evaluated at Semiscale?

ANSWER.

The Westinghouse Reactor Vessel Level Indicating System (RVLIS) has been installed on the Semiscale MOD-2A system to test its operation during a series of small break experiments. Experiment S-UT-3 was the first experiment in the S-UT series in which the Westinghouse RVLIS was installed for evaluation. This experiment was conducted to simulate a two and one half percent communicative break in the cold leg of a pressurized water reactor (PWR). Initial conditions of this experiment were typical of, or scaled from, an upper head injection (UHI) plant (although only standard plant ECC injection was used). Westinghouse supplied the instruments at no cost to the NRC and is not being charged for the tests. NRC benefits from knowing that the instruments perform their function.

The Semiscale facility is highly instrumented. Vessel instrumentation includes several coolant density and differential pressure measurements capable of determining vessel coolant inventory to a high degree of spatial resolution. The Semiscale system, therefore, represents an ideal test bed for evaluation of PWR liquid level measuring devices.

The tested Westinghouse RVLIS consists of three differential pressure measurements with connections being made at the vessel top and bottom and on the hot leg pipe. The installation at Semiscale is intended to be typical of that in a PWR; thus, the three sensors, the hydraulic isolators and the differential pressure transmitters are stated by Westinghouse to be functionally the same as standard RVLIS components and were installed and checked out by Westinghouse and Semiscale personnel in a manner typical of a PWR installation.

QUESTION 3. What design changes have caused delays at UPTF in Germany:

ANSWER.

The two major factors ultimately contributing to the delays were: 1) adding the full-size downcomer to the vessel and 2) changing the 180° -sector vessel into a 360°-cylinder vessel. The decision was reached, based, to a large degree, on a U.S. recommendation that these design changes be made. These decisions were made in early 1979.

Another reason for design delays at UPTF is that the Bundesminister fur Forschung and Technologie (BMFT) of the Federal Republic of Germany had difficulty in getting a contract agreement with its subcontractor Kraftwerk Union (KWU) on the estimated cost of building the UPTF facility. This led to delays in defining the interface and scope of supply issues, and the uncertainties in cost prediction.

At meetings held during the winter of 1981-1982 a new program plan was negotiated which allows the United States to reduce the instrumentation and analysis support in the German Upper Plenum Test Facility, and to make its costs predictable.

QUESTION 4. Describe the improved instrumentation and advanced display system to be evaluated at LOFT in FY'83. Who proposed the designs?

ANSWER.

Generally, these include the Augmented Operator Capability (AOC) displays and the technical support center.

The AOC system was designed by INEL in cooperation with reactor vendors, using results from an industrial survey performed in cooperation with EPRI. Typical instruments include an easily understood display to indicate out-of-specification parameters, a pump current/coolant temperature display for diagnosis of primary subsystem void, and a saturation pressure/temperature display which was proposed by several sources. The latter display, although not identical to ones used in industry, is connected to LOFT in the same manner and provides similar information for evaluation. These systems have been used in several LOCA and transient tests. Resulting information and conclusions have been provided to industry.

With a few exceptions that are considered unimportant for LOFT*, the Technical Support Center meets or exceeds the NRC specifications for industry. Connections to the system are similar to plant support centers. The LOFT center has some items not generally provided in industry, such as a T.V. camera that can be used to read meters in the control room, a large computer, and a direct line to the operators through a head set. Design of the center was coordinated with industry, and test experience is provided to the industry.

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* Size, satisfactory operation in respect to failures, ventilation.

QUESTION 5. What contribution is being made by Industry to the NRC work on guidelines for management of severe accidents?

ANSWER

The IDCOR program is an industry sponsored research program in the area of severe accidents. Regular NRC/IDCOR management and technical exchange meetings have been scheduled since the initiation of the IDCOR program. We continue to find the Exchanges extremely helpful in the planning of our program and anticipate being able to use the results of much of the analysis work done by the IDCOR group, just as they are using codes and phenomenological data which our program has and is developing.

Question 6

Is NRC doing any research on seismic designs which depend more heavily on the ability of large pipes to absorb energy when not held rigidly in place? Is any research at NRC addressing the tradeoffs between design and operation constraints in the use of snubbers and restraints?

NRC Response

The NRC has initiated a study in May 1982 on "Parameters Influencing Damping In Piping Systems" which seeks to relate pipe stiffness and support type to damping (that is energy absorption capacity). In addition, an effort began in April 1982 entitled "Reliability Analysis for Stiff versus Flexible Piping." Using principles of Probabilistic Risk Analysis, this effort will evaluate how design practices and conservative seismic design criteria lead to an increased number of pipe supports which stiffen piping. The research will compare and quantify, in a probabilistic sense, how the seismic conservatisms enhance safety against earthquake while at the same time reducing safety against more frequently occurring thermal transients and, in addition, diminishes the effectiveness of in-service inspection. These efforts are being coordinated with a recently formed Pressure Vessel Research Committee (PVRC) steering committee that is investigating the problem of piping systems design. This committee is composed of experts from academia, industry and the NRC who provide independent technical evaluation and advice to the ASME. NRC's principal interaction is with matters relating to the ASME Code

Question 7

Please describe the SSMRP program in more detail.

NRC Response

The Seismic Safety Margins Research Program (SSMRP) is a multiyear, multidisciplinary program whose overall objective is to develop improved methods for seismic risk assessments of nuclear power plants, using a probabilistic computational procedure. Best estimate calculations are made which include the evaluation of seismic hazard at a site, soil structure interaction, and building and subsystem response. Structural and component failure probabilities are calculated using fragility (failure vs. response level) curves and a systems (event/fault tree) analysis is made to predict probabilities of radioactive release from seismically initiated events. Uncertainties in data and modeling are considered throughout. The major contributors to seismic risk will be identified in order to focus on what licensing issues are important.

The SSMRP began in 1978 and will end in 1984. Lawrence Livermore National Laboratory has been the primary contractor to the NRC, but consultants and subcontractors have been used extensively. Earlier stages of the program dealt with methodology development. The emphasis is now on delivering usable products to the NRC. In 1983, the seismic risk analysis of the Zion plant will be completed, the methodology will be simplified into a form for generic use, and validation work will begin. A senior NRC management review group has been formed to provide overall guidance for this complex program. In addition, an external senior review group consisting of well known experts in the applicable areas provide recommendations and advise on direction of the SSMRP.

Question 8

Please describe the steel and concrete containment experiments with large penetrations being conducted under the Reactor & Facility engineering decision unit.

NRC Response

The FY 1982-83 containment integrity program is limited to understanding the behavior of steel containment structures under static overpressures. It is planned to conduct overpressure tests of six 1/32 size prototypical containment models. The last two experiments will be on ring-stiffened models including representations of major penetrations; these will provide data for evaluating three-dimensional prediction models. Previous experiments will be on unpenetrated models to provide data for comparisons of two-dimensional predictions of post-yield behavior. Finally, there will be an experiment on a model, of 1/10 size, with typical penetrations, built with conventional construction techniques.

Six test-to-failure are anticipated for concrete models in FY 1984. All models will be approximately 1/10 the size of a typical containment. The final two models will include seismic reinforcement and penetrations and will provide data against which three-dimensional predictions can be compared. The first four models will be unpenetrated, with different reinforcement patterns, to provide data for calibration of two-dimensional predictive methods.

Question 9. Describe the improved real-time ultrasonic test method under development in FY'82 and its performance characteristics.

ANSWER.

The ultrasonic test method under development is called SAFT-UT (synthetic aperature focussing technique - ultrasonic test). It depends upon computer processing and reconstruction of multiple pulse-echo signals from ultrasonic transducers, which are scanned over the component surface in order to detect flaws. The image produced through SAFT processing is not the typical line "squiggle" but rather is a visual representation of the flaw shape and location in cross section, with the capability for viewing in three dimensions. The display can be either in color or in gray scale to aid in flaw size and location perception. The method is applicable to both flaw detection and flow evaluation. Because the flaw is displayed in terms of size, shape and location, operator interpretation is eliminated.

The SAFT-UT method is nerly complete as a laboratory development. The developments to date are for image processing at a rate of 250 cubic inches of inspection volume per minute. This is an average inspection volume rate for current, automated inspection equipment. By FY'84 it is expected that the question of the practical application to full size plants of this laboratory demonstrated method will be answered, and if appropriate, submitted for consideration by the Subcommittee on Inservice Inspection (Section XI) of ASME Code.

Question 10. (a) What reactor surveillance techniques will be evaluated in FY 83?

ANSWER.

Human factors types of research to evaluate reactor surveillance techniques in FY 83 are Safety Parameter Display System (SPDS) format display research, experimental evaluation of annunciator/alarm handling and display techniques such as Halden Alarm Handling System (HALO) and development and comparison of several methods of display formats and operator response. See also response to question 32. Additional research in the area of surveillance hardware is also being performed but is not addressed here.

Question 10. (b) Who proposed these techniques?

ANSWER.

The SPDS was proposed by the NRC. HALO was proposed by the Halden Reactor Project. Methodology for analyzing display formats and operator response was proposed by Idaho Nuclear Engineering Laboratory (INEL).

Question 10. (c) What methods will be used to ensure that modifications in the operator-machine interface will actually improve safety?

ANSWER.

Both SPDS type display formats and HALO display formats are being evaluated by laboratory experiments with human operators to compare differences in performance. Several other methods of display formats that are under development are being compared.

Question 10. (d) What work will be initiated to improve quantitative estimates of human reliability?

ANSWER.

Quantitative estimates of human reliability are being verified by an ongoing research task which will continue in FY 83. Methods of obtaining human reliability data as well as the data itself will be developed and used. Data on human performance in operations and maintenance of reactor systems and components are currently being collected and assessed to identify human reliability values and ranges.

QUESTION 11. Describe NRC interaction with utilities on improvement of the impact on safety of human performance problems.

ANSWER

A coordinated research program in job/task analyses with the Institute of Nuclear Power Operations (INPO), which is a coordinating organization for the utilities, is underway. The Electric Power Research Institute (EPRI) also is a partner to the research coordination. Scheduled exchanges of information and research data takes place formally through meetings among NRC, INPO and EPRI. Informal meetings and exchanges of correspondence occurs at the working level and the management level. These types of interaction are directed toward improvement of safety through coordination of human factors research sponsored by these organizations and the NRC.

QUESTION 12. (a) Who developed the requirements for SPDS?

ANSWER

The concept for the SPDS was developed within the NRC subsequent to the TMI accident. Guidance for the concept is described in NUREG - 0696, "Functional Criteria for Emergency Response Facilities", and NUREG - 0737, "Clarification of TMI Action Plan Requirements." The SPDS is not a regulatory requirement. A staff recommendation to make it a requirement has been submitted to the Commission (SECY-82-111, "NRC Staff Recommendations on the Requirements for Emergency Response Capability") and is currently under consideration by the Commission.

QUESTION (b) Is NRC conducting any continuing research on operator information needs?

ANSWER

Research to support the identification and development of operator information needs began in FY 80, will continue in FY 82 and is planned for FY 83 and the ensuing years. National Laboratories that are currently involved in this research are INEL, ORNL, BNL, SNL. Research is also underway at the Halden (Norway) Project as well as several commercial organizations. The OECD Halden Reactor Project is a multinational program at the Halden, Norway reactor facility addressing several research areas including human factors. The USNRC is a signatory of this project.

QUESTION 13. Is NRC conducting any research on reducing the impact of maintenance on overall plant safety?

ANSWER

Yes. One program is to identify the types of likely causes of human error attributable to maintenance actions for different nuclear power plant systems, components and equipments. A second program, starting in FY 81 and continuing in FY 82, is directed toward human factors assessments of maintenance activities and maintenance personnel. In FY 81, job analyses of two maintenance positions (instrumentation and control technician and mechanical technician) were performed and a methodology for modeling maintenance personnel functions was identified. Job analyses of other maintenance positions are currently underway. Flow charting of the model has been started and software characteristics of the simulation method are being identified. Plans for FY 83 include completion of the model, job analyses for other maintenance positions and validations of the results.

QUESTION 14. "What does NRC consider to be an adequate degree of radiation protection for workers?"

ANSWER.

The NRC, in a letter to the EPA Administrator dated July 24, 1981, took the position that an adequate degree of radiation protection for workers in NRC-licensed activities will be provided if compliance with the most recent recommendations of the International Commission on Radiological Protection (ICRP Publications 26 and 30) is achieved. The following statement appears in paragraph (96) of ICRP Publication 26:

> "The Commission believes that for the foreseeable future a valid method for judging the acceptability of the level of risk in radiation work is by comparing this risk with that from other occupations recognized as having high standards of safety, which are generally considered to be those in which the average annual mortality due to occupational hazards does not exceed 10^{-4}."

This annual risk is equivalent to 4.5×10^{-3} for a 45-year working lifetime.

Approximately 167,300 workers in NRC licensed activities receive measurable annual radiation doses; the average annual dose for these workers is 0.4 rem. Using risk factors recommended by the National Academy of Sciences - National Research Council (1980 BEIR Report) it can be shown that a typical worker population, ages 20 to 65 years, each receiving an average dose of 0.4 rem per year, would be subjected to an average lifetime mortality risk of 1.1×10^{-3} for males and 1.5×10^{-3} for females, both within the ICRP recommendation.

One reason for the low average worker dose is that the NRC operates under the assumption that there is no threshold dose below which there is no risk. For this reason licensees are expected to take reasonable actions to reduce radiation doses, even those that are below NRC dose limits, so as to be "as-low-as-is-reasonably-achieveable" (the ALARA concept).

It should be noted that the ICRP recommendations include the concepts of dose justification and optimization as well as dose limitation. Optimization is an analytical method by which operational criteria can be established to minimize the total cost associated with radiation exposure, i.e., the protection costs plus the cost of health effects. These concepts are primarily applicable to the collective (manrem) dose. The collective dose received annually by workers in NRC-licensed activities is about 70,000 manrems, which may, on a statistical basis, ultimately cause about seven cancer deaths per year. The NRC believes that this risk can be reduced even further and at a reasonable cost. The research efforts planned for occupational radiation protection are intended to accomplish this, particularly for those workers who receive annual doses exceeding the average.

QUESTION 15. Please describe any research that NRC is doing to address decontamination of systems contaminated as a result of accidents.

ANSWER.

A decontamination research program initiated by the NRC in FY 82 will examine the effectiveness of available decontamination methods. This program is being coordinated with an Electric Power Research Institute (EPRI) research project which concentrates on developing methods for decontamination of reactor coolant systems contaminated as a result of accidents. The scope of the EPRI work includes: (1) review of specific accidents, such as TMI-2 and Plutonium Recycle Test Reactor (PRTR), and to review procedures and processes used in removal of fission products and fuel debris; (2) to provide reference inventory of type of fission products, their chemical forms, amounts and location throughout the reactor coolant systems; (3) determination of applicability, material compatibility, expected decontamination factors, and recognition and evaluation of any safety problems associated with various decontamination processes; and (4) evaluation of processes which might be applicable to TMI-2 accident.

The EPRI developed processes, as well as others, will be evaluated by the NRC research programs so that regulatory decisions can be made from a firm technical bases.

Question #16: Describe cooperative studies with the state of Kentucky to assist the state in decommissioning the Maxey Flats site. Is this work generically applicable?

Answer:

NRC is funding a multi-disciplinary program to develop understanding of the behavior of radioactive wastes in shallow land burial sites in humid regions. This program which is funded by NRC, involves integrated participation by the State of Kentucky, Battelle Pacific Northwest Laboratory, Brookhaven National Laboratory, Los Alamos National Laboratory, Morehead State University and the University of California, Berkeley.

Specific objectives for the Maxey Flats program include:

1. Determination of the areal distribution, types of radionuclides, and activities of contaminants.
2. Identification and quantification of those activities and processes which have resulted in release of contaminants.
3. Characterization of the trench water chemistry.
4. Characterization of groundwater flow and radionuclide transport in the unsaturated zone in terms of soil moisture changes, retardation processes, and/or changes in the radionuclide species.
5. Development of quantitative input to site performance assessment models.
6. An evaluation of alternatives for site stabilization, particularly methods to reduce infiltration of water into the trenches.
7. Development of statistically meaningful methods for monitoring radioactivity in the environment at shallow land burial sites.

The information obtained in this program will assist Kentucky in determining the extent of waste migration at the Maxey Flats site; it will also produce practical generic information needed for developing criteria for managing, closing, and monitoring other shallow land burial sites in humid regions.

Question #17: What engineering designs for managing uranium mill tailings are being evaluated by NRC? Who proposed them?

Answer:

The research program on management of uranium mill waste is testing the effectiveness and reliability of engineering methods for managing active uranium mill tailings and retired tailings piles relevant to protection of the public from hazardous levels of radioactivity. The results of this research will be used to provide technical bases for guides and to support licensing decisions.

Research on active tailings includes the following three studies: 1) An assessment of the effectiveness of methods for immobilizing toxic and radioactive materials in tailings and thereby mitigating seepage into groundwaters. 2) An assessment of the effectiveness, benefits, and costs of tailings dewatering techniques for reducing tailing impoundment seepage losses into groundwater. 3) An assessment of the effectiveness and durability of interim stabilization techniques for reducing the dispersion of particulates from exposed active tailings surfaces.

Research on the retired tailings includes the following studies: 1) Testing of rip-rap and other materials to confirm their capability for providing long-term protection of the tailings from water-and wind-erosion and dispersal resulting in public exposure to radionuclides, 2) Testing of the effectiveness of various methods and materials for capping tailings to attenuate radon emissions. These studies will provide NRC with information and reliable models for evaluating proposed covers and to verify the effectiveness of in place covers.

A small part of the research program is devoted to studying alternatives to surface disposal. This effort currently includes assessments of engineering methods which would involve tailings below grade in saturated media.

The engineering designs and methods under investigation have been selected for study on the basis of licensing experience, DOE's experience and the research staff's knowledge of existing problems and the developing state-of-the-art.

QUESTION 18. Please describe NRC research programs to protect "the health and safety of the public and workers from potentially harmful effects... of ionizing radiation and radioactive materials." (NRC Budget Estimates - Fiscal Year 1983, p. 75)

ANSWER.

The health effects research program limits its support to studies that have more immediate application to regulatory requirements or questions. This means that the NRC must rely on the larger biomedical research programs of the National Institutes of Health and the Department of Energy, and other national and international sources of information to provide the necessary scientific base for our radiation protection decisions. The annual research budget of the NRC constitutes only about 3% of the total Federal research budget devoted to biological effects of ionizing radiation. Obviously, the NRC could not perform its radiation protection mission without the information provided by this broader base and we are working with the present Interagency Radiation Research Committee to improve this necessary interchange.

During FY83 the following projects are to be funded:

"Health Effects of Industrial Exposure to Thorium"
Argonne National Laboratory

Former workers at a thorium processing plant are being followed to determine if exposure to thorium has caused increased respiratory illness or cancer deaths. Deposition of thorium and its daughter radionuclides in body tissues is being examined by external measurements and by analysis of autopsy samples. The results will be used to assess the adequacy of the present limits for the maximum permissible concentration of thorium in air for occupational exposure.

"Relative Biological Effectiveness of Fission Neutrons and Gamma Rays at Occupational Exposure Levels"
Argonne National Laboratory

A large population of mice is exposed weekly to either fission neutrons or gamma rays at doses comparable to the current occupational limits; tumor incidence, life-shortening, and genetic effects will be compared with a control population. The results will be used to evaluate the relative biological effectiveness of neutrons and determine if the present limits for occupational exposure to neutrons are adequate.

"Radiation Dose Estimates and Hazard Evaluations for Inhaled Airborne Radionuclides"
Lovelace Inhalation Toxicology Research Institute

Rats, dogs, and monkeys are exposed by inhalation to mixed uranium and plutonium oxides collected from industrial facilities; patterns of deposition and dose-effect studies are conducted. The results will be used to improve calculation of doses to organs from inhaled materials and to assess the adequacy of present limits for maximum permissible concentrations in air for occupational exposure.

"Biological Characterization of Radiation Exposure and Dose Estimates for Inhaled Uranium Milling Effluents"
Lovelace Inhalation Toxicology Research Institute

Rats, dogs, and possibly monkeys are exposed by inhalation to various yellowcake products collected at operating uranium mills; patterns of deposition and dose-effect studies are conducted. The results will be used to improve calculation of doses to organs from inhaled materials and to assess the adequacy of present limits for maximum permissible concentrations in air for occupational exposure.

"Early Morbidity and Mortality for Different Nuclear Accidents"
Lovelace Inhalation Toxicology Research Institute

and

"Early Effects of Inhaled Radionuclides"
Pacific Northwest Laboratory

In this study the effects of high level exposure to a variety of radionuclides with and without external exposure are being investigated. The study is done in two laboratories, each performing the part for which it has the greatest expertise. The objective is to provide experiemental data necessary to develop models for mortality and morbidity which would be expected to occur within 18 months after exposure to various radionuclides. This information will improve the estimates of potential health effects of postulated accidents in nuclear power plants, fuel cycle facilities and transportation.

"Biodosimetric Confirmation of Dose Rate Amelioration Factors"
University of California at Davis

The NRC sponsored study is a part of a larger study supported by DOE. Dogs are being continuously irradiated at different dose rates and their hematologic and immunologic status monitored. The NRC is funding the low exposure rate part of the study. The objective is to determine if early changes in hematologic and immunologic systems can serve as predictors of late effects (cancer) and as biodosimeters, and to determine quantitatively the difference in effectiveness of equal doses delivered at different dose rates. This study will contribute to determination of health effects from exposure to low-level ionizing radiation and as such have an impact on radiation protection standards.

"Follow-Up Study of Patients Who Had I-131 and Other Diagnostic Procedures During Childhood"
Bureau of Radiological Health (BRH), FDA

In this study, jointly sponsored by the NRC, BRH and NCI, persons who received diagnostic I-131 before age 16 and others who received a different thyroid test are being followed, and their medical data reviewed and analyzed; in order to determine if the exposure to diagnostic levels of I-131 is associated with increased incidence of thyroid cancers and benign nodules. The results of the study will help to reduce uncertainties in risk estimates of malignant and benign thyroid neoplasia resulting from exposure to I-131.

QUESTION 19. Please describe in detail work to be funded in the Systems and Reliability Analysis decision unit. How are the results of this work being used by the licensing and inspection staff in prioritizing and resolving safety issues? Please cite examples. How does this work help to stabilize regulatory practices? Please cite examples.

ANSWER.

In addition to Interim Reliability Evaluation Program (IREP) and National Reliability Evaluation Program (NREP), FY 1983 activities will develop value/impact methods for staff assessments of the effectiveness of proposed changes in regulatory requirements and/or proposed backfitting actions. Studies of accident sequences and licensee event reports will provide an improved framework for evaluating the effectiveness of our current nuclear power plant regulations and for developing effective inspection protocols. High priority will be accorded to developing methods for assessing common cause failures such as fires.

The methods, techniques and insights developed in the Systems and Reliability Analysis (SARA) program over the last several years are being used to support Commission decisions in a variety of areas. NRC has employed SARA-developed techniques to aid its regulatory decisions on the turbine missile question, the ATWS issue (Anticipated Transient Without Scram), assessing the reliability of auxiliary feedwater systems and the implications of a loss of offsite power on nuclear power plants. Current applications of SARA-developed Probabilistic Risk Assessment (PRA) techniques are in the implementation of the Commission's Safety Goal, the development of regulations dealing with Severe Accidents (e.g., hydrogen control, decay heat removal systems, etc.), direct support for the Commission's policy on siting and emergency planning, and the Commission's efforts on IREP and NREP.

The largest single activity in Systems and Reliability Analysis is the IREP/NREP activity. This work is directed toward the development and trial use of cost-effective methods to perform specific risk analysis of plants. The results and methods developed are being used directly by the licensing office. For example, results of the Reactor Safety Study Methodology Application Program (RSSMAP) has been used for the following purposes:

1. Indian Point interim operation decisions;
2. Environmental Impact Statements;
3. Considerations of hydrogen control systems for ice condenser containments; and
4. Considerations of alternative designs for severe accident prevention and mitigation.

See also the response to question 21 for additional areas where PRA has been used in the licensing process.

Probabilistic Risk Assessment (PRA) techniques are being used as an aid to prioritizing research by permitting the output of such research, (either in terms of better data on physical processes or more realistic analytical models) to be compared on the basis of their effect on risk reduction. The Office of Nuclear Reactor Regulation (NRR) is using PRA to prioritize their list of unresolved safety issues and RES is using PRA to provide a better focus to its program.

QUESTION 20 Please describe NRC's operations research as applied to licensee safety assurance practices.

ANSWER.

The NRC is currently engaged in a program to evaluate the risk limitation effectiveness of a systematic reliability assurance approach to reactor safety. This program is a comprehensive, multidisciplinary effort, aimed at the detection and rectification of hidden hardware or procedural deficiencies that might have been overlooked during the design phase or missed by the PRA, QA/QC and surveillance testing; it will explore the benefits of integrating and adopting aerospace and other high technology reliability systems engineering disciplines and techniques into and by the nuclear field. This action is responsive to Section II C4, NUREG-0660, NRC Action Plan Developed as a Result of the TMI-2 Accident. The current effort is focused on the development of a pilot reliability assurance program limited to addressing problems associated with anticipated transients without scram (ATWS). The cost-effectiveness of this safety assurance approach will be evaluated as part of the pilot study.

Please see the response to question 22 for a related program effort.

QUESTION 21. Are there any results to date on analyses of the regulatory structure and safety evaluation practices? If so, please cite examples. Please explain the term regulatory structure as used on page 75 of NUREG-0870.

ANSWER.

There are only limited results available to date on the analysis of regulatory structure and safety evaluation practices. By regulatory structure, we mean to embrace both the general structure, logic, and consistency of our regulations and the embedded regulatory practices. An often-cited example of the latter is the Design Basis Accident method by which plant safety is conventionally analyzed. Of its very nature, a probabilistic risk analysis (PRA) is an alternative method to check the effectiveness of the DBA method.

Probabilistic Risk Assessments, which attempt to treat realistically multiple failures and a wide range of possible accident scenarios, have in the past shown gaps in existing regulations. For example, following the TMI accident, probabilistic analyses were performed on the auxiliary feedwater systems of each operating PWR. Based on these reviews, numerous modifications were required in order to upgrade the reliability of these systems. Our Standard Review Plan now incorporates instructions for a supplementary PRA analysis in addition to the standard safety evaluation practices. Similarly, PRA results have been used to prescribe improvements in the reliability of onsite AC and DC electrical power systems, and to permit reduction in the likelihood of a potentially serious accident in PWRs--the "interfacing systems" loss-of-coolant-accident.

In addition to the above, a small study (Coherence Analysis of 10 CFR Chapter I) was completed in 1981 to analyze the structure of certain of NRC's regulations. The study was preliminary in nature; its purpose was to determine if a technique known as the issue tree method could be used to examine the structure, logic, and consistency of selected parts of 10 CFR Chapter I. The results are being used to assist the NRC staff in conducting the periodic and systematic review of the regulations.

Question 22. What programs are planned or underway to investigate alternative techniques of system safety assurance drawn from other industries? Please describe results to date.

Answer.

An Interagency Agreement between NRC and NASA-Kennedy Space Center (KSC) is about to be signed to investigate applying the NASA/KSC methodology of System Assurance Analysis to nuclear plant-specific safety systems. The goal is to develop guidelines (e.g., handbook, design manual, etc.) for its application to nuclear system safety/reliability analyses.

QUESTION 23. What improvements were suggested for B&W auxiliary feedwater systems using risk analysis techniques in 1980?

ANSWER.

The following is a brief listing of the recommendations made by the Office of Research to the Office of Nuclear Reactor Regulation (NRR) regarding steps which warranted consideration for B&W auxiliary feedwater systems. These recommendations are being implemented by NRR.

a. Modify the Technical Specifications to limit the time that one emergency feedwater system pump and its associated flow train can be inoperable.

b. Lock open manual valves which, if closed, could interrupt all flow in the system. Verify they are locked in the proper position at least monthly.

c. Reexamine the practice of throttling flow to avoid water hammer. Verify the system will provide sufficient initial flow to assure adequate decay heat removal following a loss of main feedwater.

d. Emergency procedures for transferring to alternate sources of water should be available.

e. The plant should be capable of providing the required emergency feedwater system flow for at least two hours from one pump train, independent of any AC power source. Procedures should be available for manually initiating and controlling the system under these conditions.

f. The licensee should propose Technical Specifications to assure flow path availability of a flow train when restoring the system to service after periodic testing or maintenance.

g. The licensee should verify that the automatic start signals and associated circuitry are safety grade or modify it to meet certain criteria.

h. The licensee should install a system to automatically initiate emergency feedwater system flow.

i. The licensee should provide redundant level indication and low-level alarms in the control room for the primary source of water.

j. The licensee should perform a 72-hour endurance test on all emergency feedwater pumps.

k. The licensee should provide safety grade instrumentation, powered from the emergency buses, which will indicate in the control room the emergency feedwater system flow to each steam generator.

l. Plants which require local manual realignment of valves to conduct periodic tests and which have only one remaining train available should provide a dedicated individual, in communication with the control room, be stationed at manual valves.

m. The licensee should assure plant procedures are written to reduce human-induced errors. The licensee should implement staggered testing of emergency feedwater system trains.

n. Information should be provided to the operators on means to detect and cope with emergency feedwater interruptions caused by failures in the Integrated Control System or the Steam and Feedwater Detection and Mitigation System.

o. The licensee should assure that there are no temporary strainers in the system.

p. Plants should be evaluated to determine if automatic protection of the pumps is required to prevent damage resulting from inadequate net positive suction head caused by a seismic event or tornado.

Similar suggestions were developed for feedwater systems in Westinghouse and Combustion Engineering plants.

Question 24. Describe the NRC efforts to "identify off-target or unnecessary regulations" using Systems and Reliability Analysis. What regulations are under consideration in this work?

Answer.

A study to examine the effectiveness of certain regulations in limiting public risk is planned for FY'82 and 83. The study will be similar to the 1978 study to rank 133 generic safety issues dealing with nuclear power plants. Only 15 issues in that study were found to be important to safety. The basis for the new study will be the nuclear power reactor probabilistic risk assessments (PRA) and safety system reliability assessments that have been and are currently bein published. These PRAs contain many findings about the comparative importance to safety of system design features and operations practices. The new study is expected to identify and document any evidence supporting inferences of overregulation as well as cases of safety-significant loopholes in the regulations.

The study will concentrate on appropriate portions of the General Design Criteria for Nuclear Power Plants (10 CFR 50, Appendix A), appropriate parts of 10 CFR 50, selected Regulatory Guides, and portions of the Standard Review Plan. Some current activities in developing new siting regulations have already identified details of engineered safety features design and review practices which appear to be unnecessary. These relate to containment building leak rate testing stringency and containment atmosphere cleanup systems.

Question 25. How do NRC researchers develop their expertise on reactor operations? What percent of the research staff has utility experience? Reactor manufacturing experience?

Answer.

The major portion of the research staff received their initial training prior to joining the Office of Nuclear Regulatory Research. This is supplemented by specific training received in accordance with Individual Employee Development Programs which are required for each employee.

The following breakdown represents the experience and backgrounds of the research technical staff. Many staff workers have past experience in multiple categories but for purposes of this display only their most relevant experience was considered. Because NRC records do not lend themselves to the rapid compiling of this type of data, it was obtained by polling the first line supervisors and therefore, should be considered as only approximate in nature.

Utilities 3%	Vendors 10%	Architectural Engineering 7%	
Naval Reactor 4%	Laboratories (Nuclear Related) 19%	DOE Nuclear Programs 10%	
Other NRC Offices 20%	New Graduates (within 2 years) 13%	Consultants 10%	Other 4%

Question 26. Describe the process used by the NRC research organization to set priorities for the use of research dollars. To what extent are relative risk comparisons used in establishing the research budget request? Cite examples.

Answer.

A multi-step process of program review, comment and endorsement has been established to develop the NRC research program. The first step in the sequences is the development of the NRC Long Range Research Plan which is based on: (1) comments from the prior plan; (2) specific program and planning guidance received from the Commission; (3) needs based on requests from the Office of Nuclear Reactor Regulation, Office of Nuclear Material Safety and Safeguards, and the Office of Inspection and Enforcement; (4) problems identified by previous research; and (5) incorporation of ACRS comment contained in their report to Congress. This plan is intended to provide broad program direction for the NRC research effort.

The Long Range Research Plan is then reviewed by the ACRS and user offices. All comments are considered and the plan is revised. User office endorsement of the plan is then requested and the plan is submitted to the Commission for concurrence.

The Long Range Research Plan provides broad program direction and a basis for detailed program formulation by the research staff for the budget year. The budget is then reviewed and commented on by user offices, ACRS, and the Commission. This iterative process is the structure in which research priorities are established. A significant input in the setting of priorities is a qualitative judgment of perceived high risk or high risk uncertainty.

Specific examples of areas which have had significant resource cutbacks in order to fund what are judged to be higher priority projects are: safeguards research, environmental impact research, large break LOCA/ECCS research and research to confirm margins inherent in existing regulations.

Areas of increased emphasis based on uncertainty and high possible risk are: pressure boundary safety, operability of equipment, integrity of electrical connections, structural integrity of aging plants, assessment of nondestructive examination techniques and application of risk assessment techniques to better focus the regulatory process.

QUESTION 27. Are annual and monthly routine reports described in Regulatory Guide 1.16 used in setting research priorities? What analysis of these data is conducted by NRC research? What about analysis and use of reportable occurrences described in the same source? For instance, are causes of potentially serious incidents analyzed for trends? What percent of potentially serious incidents show human action to be a direct or collateral cause? How often does incorrect or tardy maintenance contribute to a potentially serious incident?

ANSWER.

The annual reports are not a prominent source of information used in setting research priorities. The reportable occurrence and monthly reports have significant use in calculating the failure rates of safety systems and components in safety systems. Component failure rates developed from these reports are used in the estimation of plant systems unreliability. In addition, components critical to safety are analyzed using this data to determine if their failure rates are increasing or decreasing with time.

The frequency of forced plant shutdowns and component failure rates derived from the three report types are used in the Probabilistic Risk Assessment (PRA) of safety of nuclear plants. These analyses, within the extent of PRA accuracy, highlights those aspects of plant equipment or operation which can significantly affect risk.

It should be noted that the data base used in reliability assessments and PRA's is not limited to data provided by licensee reporting. Such data are complemented by information developed in other areas of the research program such as in the areas of code development and verification and the analysis of transients.

Reportable occurrences are being analyzed for potential safety impact in a research program called the Precursor Program. This program is an attempt to more directly utilize operational experience data in assessing the risk from commercial nuclear plants. The precursor program models each potentially significant abnormal occurrence to show how close the reactor came to predicted severe core damage because of the incident. The Precursor Program uses system level operational experience data directly in assessments rather than synthesizing more basic component data mathematically by fault trees. The precursor program will eventually provide a measure of plant risk which may corroborate the accuracy of current PRAs. The precursor program shows a significant fraction, (approximately 30 to 40 percent) of the severe core damage probability is contributed by human error.

The abnormal occurrence reports of incorrect maintenance are of considerable interest because of the potential for common cause failures from incorrect maintenance. Tardy maintenance is usually of minor interest because there is normally a fairly large range of maintenance intervals which cause a relatively minor variation of component or system unavailability. Although we do not have sufficient analysis yet in hand to quantify it, incorrect maintenance appears to be a substantial contributor to severe accident probability.

Question 28. If the NRC were to reopen proceedings related to the Generic Environmental Statement on Mixed Oxide Fuel (GESMO), would additional research be required? If so, please describe the scope and magnitude of that research.

Answer.

The answer is dependent upon the scope of the proceedings. If this scope is restricted to the LWR Fuel Cycle, then it is the staff's opinion that there is need for additional research in the area of safeguards. However, if the scope includes recycle of the mixed oxide in reactors other than LWRs, i.e., breeder reactors, then additional research would be required in all areas. Our estimates of the broad areas requiring additional research are:

Safeguard Research

Additional safeguards research would be required. This research could be expected to address the problems of safeguarding the additional quantities of plutonium or plutonium containing materials which would be introduced into commercial operations.

The increased safeguards concerns about plutonium arise from its potential use in nuclear explosives devices and its radiological toxicity. The quantities of these materials now in the licensed industry are limted to those present in spent fuel of light water reactors, and to those used for R&D purposes. Nuclear fuel reprocessing, possibly coupled with commercial licensing of breeder reactors, would result in the introduction of significant quantities of plutonium in more accessible form, in commercial operations.

The priority areas where additional research would be required are:

- Tracking of plutonium bearing materials in transit
- Development of a system for real time nationwide communications with transit shipments
- Material control in large scale reprocessing and mixed oxide fuel fabrication plants, to include applications of discrete process unit control and accounting
- The adequacy of current physical protection rules and the proposed Material Control and Accounting Reform Amendments for reprocessing plants

Question 28

- Defining an effective network of resident inspectors for a large-scale reprocessing plant
- Integration of the Physical Protection and Material Control and Accounting Systems to optimize the use of resources and minimize the impact on licensees
- Development of material control alerting signals for the Physical Protections System

Safety Research (for advanced Fuel cycles only)

1. Fast Reactor Safety

   Analysis of core-melt accidents

   - SIMMER code validation through experimental data (expensive work)
   - Analysis of breeder core's ability to cool through natural convection
   - Reliability of decay heat removal systems
   - Analysis of re-criticality risks while fuel is molten after a breeder core meltdown

2. Fuel Cycle Safety

   Determination of isotopic source terms for accident analysis

   - Different proportions of fissile isotopes may increase prompt criticality danger in fuel processing operations
   - Isotope source terms for environmental dispersion analyses

     What aerosol source terms?

     Groundwater solubility and transport estimates

   - Criticality analyses of aqueous systems in fuel reprocessing and recycle operations

3. Transportation

   Criticality analyses

   Dose estimates

   Detection technology

Question 29. (a) Why is the FY 83 Safeguards Program based on the assumption that nuclear fuel reprocessing will not be proposed by the industry?

(b) What adjustments or additions would be necessary if the assumption were removed?

Answer.

(a) When the FY 83 Safeguards Program was prepared there were no pending requests for a license to construct a reprocessing facility. Neither then, nor now, are there any indications that any request for a reprocessing license is imminent. The assumption that reprocessing would not be proposed in the near future by industry permits NRC to use its limited research funds in higher priority areas.

(b) If indications should develop that a reprocessing facility license application might be received in the near future, the projected research program for FY 83 would have to be adjusted. Reprocessing would introduce much larger quantities of plutonium in an accessible form into operations which are NRC regulated. At present, only the plutonium in spent LWR fuel and the small quantities of plutonium in research and development applications are regulated by the NRC.

A significant impact on our research program would result if a mixed oxide fuel cycle was anticipated. This would involve additional effort for fixed sites such as reprocessing and fabrication facilities in developing criteria to integrate material control and accounting measures into the process designs and consolidating this with physical security measures.

The development of a satisfactory transportation system including physical security, in-transit tracking and communications for large quantities of strategic special nuclear material would also be necessary.

Question 30. (a) The February draft of the Long Range Research Plan for FY 84-88 states that "Human performance is now generally recognized to be a significant and perhaps dominant determinant of the risk resulting from operating nuclear facilities." How much of the NRC research budget directly addresses this problem?

Answer.

The FY-82 research budget for human engineering in the Facility Operations Decision Unit is $4.2M and the human reliability share of the Systems and Reliability Analysis Decision Unit is $1.3M. Both are used to support research to provide data, information, methods, and technology bases to address questions on human performance.

Question (b) Please describe the program elements included.

Answer.

The four major program elements supported by the combined funds of $5.5M.

1) Human Factors Systems Engineering - this effort generates information, data, methods and standards relevant to evaluating the operator-machine interface of nuclear facilities.

2) Licensee Qualifications - this element generates information, data, methods, and standards relevant to evaluating the training and licensing of plant personnel and the management of design, construction and operation of nuclear facilities.

3) Plant Procedures - this effort addresses research and standards relevant to developing and implementing sound procedures.

4) Human Reliability - this effort develops and verifies models of human performance and quantitatively assesses the human contribution to risk.

Question 31. Please describe the respective roles of NRC and DOE in nuclear safety research. What is the nature of NRC's participation on the DOE working groups formed in response to P.L. 96-567?

Answer.

The Nuclear Regulatory Commission is cooperating fully with the implementation of this legislation and actively participating in the various DOE/NRC/industry working groups set up to work on specific sections of Public Law 96-567. These working groups consist of experienced senior technical personnel from NRC, the Electrical Power Research Institute (EPRI) and other nuclear industry user's groups and associations.

RES reviewed the draft documents prepared by the Department of Energy (DOE) in response to Public Law 96-567. Specifically, NRC agreed with DOE recommendations to forego the building of a National Reactor Engineering Simulation Facility and forego the creation of a Federal Nuclear Operations Corps.

NRC also provided inputs and comments to DOE's Program Management Plan for the conduct of a Research Development and Demonstration Program for Improving Safety of Nuclear Power Plants. Specifically:

- NRC supports the Probabilistic Risk Assessment Program proposed by DOE.
- DOE should consider initiating some of the programs proposed under the Improved Reactor Safety Program; namely:

  Add-on decay heat removal systems

  Vented Filtered Containment

  Operator-Machine Interface

  Advanced Seismic Designs
- DOE was urged to place a significant effort to implement Section 4(a)4 of the Act which calls for identifying the effects of total or partial automation of generic plant systems on reactor safety, operation, reliability, economics, and operator performance.
- Maintain the status-quo position between DOE and NRC with regard to the study of fuel failure at higher than standard burnup levels and with regard to fuel-cladding interactions. However, DOE should be prepared to do additional work in these areas if neither NRC nor industry is prepared to do it.

  NRC suggests that DOE prepare for this by budgeting for maintenance-level financial support of facilities such as PBF in which the back-up work would be done.

Question 31.

- NRC will cooperate and encourages DOE to conduct experimental and analytical research in thermal hydraulics toward development of higher reliability and reducing the cost from accidents.
- NRC supports the implementation of a coordinated hydrogen behavior and control program by DOE.
- NRC provided DOE with detailed comments on DOE implementation proposals of Section 4(a)(5)(H) of the Act dealing with Fission product release and transport from failed fuel. It was pointed out that it is more important that research on accident source terms be considered in a broader context than magnitude alone. In this area, NRC stands ready to work with DOE in better defining the parameters and constraints.
- NRC is aware of and encourages the current efforts underway to identify the aptitudes, training and manning levels which are necessary to assure reliable operator performance under normal, abnormal, and emergency conditions (Section 4(a)(7) of the Act).

A major challenge of safety research is to provide improvements in individual safety areas without creating problems for other parts of the system. The proposed DOE Program Management Plan for improving the overall safety of light water reactor reactors, which is NRC's mission, should provide an adequate balance between finding solutions to detailed technical problems and making sure those solutions provide a significant improvement to overall reactor plant safety.

QUESTION 32 Is NRC research currently addressing questions involving the use of computers at the primary operator-reactor interface? If so, describe the research. If not, why not?

ANSWER

The research projects related to human factors of display for SPDS are mentioned in the answers to Questions 12 and 30. The specific objectives of these projects are:

(1) to analyze new man-machine operational display concepts and the effectiveness in enhancing operator capability and

(2) to develop human factor display evaluation methods and evaluate their effectiveness in enhancing operator capability.

Using the results from these projects, NRC is undertaking research at INEL into the computer hardware and software requirements necessary to achieve such display methods. Some of the issues to be addressed by this project are:

- Required redundancy and diversity of computer hardware and software
- Verification and validation of the computer software programs
- Reliability of various computer architectures
- Data and Information management to achieve the required display method
- Seismic qualification of equipment.

This research project will develop the evaluation criteria needed in the licensing process of reviewing SPDS designs using computers which could come from as many as 25 different vendors. These criteria are scheduled to be published by 12/31/82.

QUESTION 33. Is NRC currently developing methods for judging the competency of a utility, its management, and personnel?

ANSWER

Yes. Research to develop a program of management analysis to support safety was begun in FY 82, will continue in FY 83 and be completed in FY 84. A task analysis of supporting personnel functions, including management tasks and functions, is part of the research. Other human factors research, which began in FY 80 and continues in FY 82, is directed toward safety related operator performance effectiveness assessment and will continue in FY 83. See response to question 30 regarding other areas of research for judging the competence of plant personnel.

Question #34: What is the breakdown, in terms of money and manpower, of the waste management program for (a) commercial high-level waste, (b) commercial low-level waste, (c) waste from uranium recovery operations, and (d) any other waste management R&D programs?

Answer:

The breakdown of the waste management research program in the NRC's Office of Nuclear Regulatory Research is as follows:

	FY-1983	
	Budget ($Million)	Manpower (Staff Year)
High-Level Waste	6.1	12
Low-Level Waste	4.3	8
Uranium Recovery	3.2	6
Total	13.6	26

Question #35: In uranium tailings research, is probabilistic risk analysis used in assessing the alternatives to covers for attenuating radon emanations?

Answer:

Probabilistic risk assessment is not being used in this program. Except in the very near vicinity of a pile, radon emanations from the pile will not result in increased levels that can be differentiated from the natural background levels of radon. Any calculated uncremental risk to an individual at greater distances is, of course, small, decreasing as a function of distance from the pile. One can calculate the total impact in terms of risk to a large population by assuming an average exposure for an individual in that large population, the so-called total population risk, but this concept can be misleading in that the risk to the majority of individuals will be so small that it is not likely to be of concern to the individual exposed.

NRC is supporting a research program to test and evaluate various earth covers designs and materials which attenuate radon emission by taking advantage of geochemical retardation and the short half-life of radon-222 (about 3.8 days). The data on reduced radon emission obtained from the tests will serve as indicators only of the relative risks associated with individual cap configurations since no other variables are being examined.

Question #36: What is the nature of the present interaction with EPA personnel? How do you propose to improve the interagency cooperation between NRC and EPA? Likewise, DOE and NRC?

Answer:

Since the programs in waste management, fuel cycle, and safeguards are closely interrelated between DOE and NRC, the Office Director of Nuclear Material Safety and Safeguards (NMSS) and his Division Directors meets frequently with equivalent level DOE management to discuss areas of mutual interest. NMSS meets somewhat less frequently with EPA management the latest meeting between the Director of NMSS and the Assistant Director of EPA's Office of Air, Noise, and Radiation was held on February 25, 1982 to discuss items of mutual interest including schedules for publication of the EPA generally applicable environmental standards for waste management and fuel cycle programs.

In the area of radioactive waste management research, NRC's research office maintains both formal and informal contacts with both EPA and DOE personnel and their contractors. Cooperative exchanges are carried on among the three agencies, however the NRC contacts with DOE are more frequent and extensive due to the size and direct relevance of the DOE waste management research program to NRC. Frequent informal exchanges are accomplished through site visits and interactions at meetings and symposia on radioactive waste management where NRC, EPA, and DOE personnel and their contractors are in attendence. In addition to these interactions, the three agencies routinely exchange technical reports which they or their contractors have produced.

Question #37: How do you incorporate the results of research by other agencies particularly DOE and EPA, or other research organizations, in high-level waste program?

Answer:

The NRC RES staff and contractors are following the progress of research program sponsored by DOE, EPA, EPRI, foreign governments, and international agencies in order to avoid unnecessary duplication and promote maximum utilization of existing information in conducting the NRC research program. The R&D program that is sponsored by DOE is the most extensive and serves as a primary source of information which is assessed by the NRC and which is evaluated to help identify issues, potential problem areas and priorities of the NRC research program.

Question #38: What are NRC's research activities regarding transuranic waste in the range of 10-100 nanocuries?

Answer:

The Commission is completing action on 10 CFR Part 61 which will establish an upper bound on the concentration of transuranic and other radionuclides in waste acceptable for near surface disposal. As part of this effort we are specifically examining all available information concerning transuranics in the 10-100 nanocurie per gram range.

As part of this effort, NRC analyzed the disposal of a broad range of waste streams, including several containing transuranic radionuclides, such as power reactor wastes. To assist waste generators in meeting these Part 61 Waste Classification requirements, NRC is developing further data to relate the concentrations of easily measured radionuclides (i.e., those which emit readily detectable gamma radiation, such as CO-60) with those that are difficult to measure (i.e., those which emit less penatrating alpha or beta radiation, such as transuranics). Such data would permit use of scaling factors and thus provide one way power reactor licensees can determine the concentrations of difficult to measure radionuclides through comparison with concentrations of radionuclides that can be easily measured.

Question #39: What is the status of the NRC effort to develop models for estimating quantitatively the performance of high-level waste repositories?

Answer:

As part of the development of a risk methodology for HLW, NRC has supported the development of mathematical models of heat transfer groundwater flow, solute transport (for repositories having salt as a host material), and radionuclide transport for repositories having a wide variety of host materials. These models are applicable to repositories of unreprocessed spent fuel as well as repositories of HLW created by reprocessing.

In addition to the groundwater transport models which have been developed as part of NRC's HLW risk methodology program, NRC is supporting research on the flow of groundwater through fractured media and plans to support research on the modeling of geochemical retardation of radionuclides from HLW. Since models of radionuclide release and transport within the engineered facility and near-field will be very design specific, the NRC will depend largely on DOE research and design efforts for the performance evaluation. Research by NRC will be done to provide the capability for independent NRC assessment of uncertainties.

Question #40: Would NRC need to do additional research regarding the disposal of high-level waste created by reprocessing? How long would such R&D take to complete?

Answer:

The current NRC waste management research program encompases the geologic disposal of solidified high-level wastes generated from reprocessing of spent nuclear fuel. Therefore, no significant additional research is contemplated currently.

Question #41: In the Review and Evaluation of the Nuclear Regulatory Commission's Safety Research Program for fiscal year 1983, the ACRS stated "the DOE schedule for site selection and repository construction may be accelerated, perhaps by as much as 2 years. Final DOE site recommendations may therefore occur as early as January 1983. In order to evaluate properly the recommended sites, the NRC should have final siting criteria prior to that time. While the proposed FY 1983 budget for this Decision Unit will permit the NRC to meet this schedule. It will necessitate some curtailment in the level of effort originally planned." Does this reference to a site recommendation mean site characterization? In reference to "curtailment" and "level of effort originally planned," what is the original schedule for this research program? In addition, the same report goes on to state, "As a result, the NRC staff will have a reduced level of confidence that the criteria they develop will assure an acceptable disposal facility." What won't be done under the accelerated schedule to result in such a loss of staff confidence? What level of confidence does the current program assure? Is this adequate? If this is not adequate, what is needed to assure confidence?

Answer:

The ACRS apparently used the term "site recommendation" to refer to the submission of a site characterization report to the NRC for comment. The high-level waste Technical Rule (10 CFR 60) dealing with technical criteria for siting and other aspects of deep geolcgical disposal is planned to be in place as an effective rule before January 1983. Clearly, the FY83 research program will have little effect on the criteria. On the other hand, the NRC technical criteria together with the selection by DOE of specific sites for characterization can help to focus the NRC research program on more specific issues relative to the regulation process.

In reference to "curtailment" and "the level of effort originally planned," earlier research planning included work that indicated a larger FY83 budget level than has now been proposed. The FY83 budget estimate reflects a reduced program of in-situ testing and essential elimination of the repository design and engineering studies that were earlier planned. NRC staff have concluded that the design specific character of such research makes it more appropriate for NRC to rely largely on DOE research and design efforts. Although this somewhat reduces the level of independence in the NRC assessments of the DOE Site Characterization Report and the construction permit and operating license submissions, we believe the current program together with prudent review procedures will provide an adequate basis for NRC decision making.

Question 42. Please describe the allocation of the $13 million request for advanced reactors for FY 1983. Why do these two new reactor designs require relatively little funding in comparison with the better known light water reactors?

Answer.

$2.5 million is allocated to safety research for high temperature gas cooled reactors (HTGRs) and $10.5 million for safety research or liquid metal cooled fast breeder reactors (LMFBRs).

A significant body of research applicable to HTGR licensing has been carried out by the Department of Energy (DOE) and the industry. The NRC effort devoted to a geneneric HTGR program is focused on the pre-application review of proposals for a prototype plant and on gathering data crucial to early licensing decisions on such an application, should it develop.

LMFBR funds will be directed at research to support the licensing review of the CRBR reactor. A substantial body of research applicable to CRBR licensing was carried out under prior programs. Major items are development and application accident evaluation computer codes such as: SIMMER for core disruptive accident, SUPER SYSTEM CODE for systems analysis, COMMIX for thermal hydraulic analysis, and CONTAIN for containment analysis. Experimental work on sodium-concrete interaction and accident energetics will continue.

QUESTION 43 Is there any chance that research needs for CRBR will delay the regulatory review and licensing of that plant?

ANSWER

At this time neither NRC research (RES) or the reviewers in the CRBR program office of NRR forsee research needs delaying the review and licensing of the plant.

A substantial body of research applicable to CRBR licensing was carried out under prior programs. Although some additional research will be required to support CRBR review, at this time neither the Research nor Licensing staffs of the NRC foresee the unavailability of specific research results delaying the review and licensing of the plant.

QUESTION 44 Describe the interface between DOE and NRC research for CRBR.

ANSWER

In FY 1983 the entire LMFBR research program is directed at providing information needed for the CRBR construction permit and operating license reviews. In developing this program, our licensing staff and our research staff have had numerous discussions with DOE to assure that our programs are coordinated and that all the information necessary for licensing will be available. The NRC licensing staff and representatives of DOE and their contractors hold frequent meetings (4 or 5 per month) for detailed discussion of issues in the CRBR license application. In this review process the need for additional research information is quickly identified and discussed by the NRC and DOE staffs.

QUESTION 45. What changes have occurred in the FY 1983 program in response to the ACRS review?

ANSWER.

With few exceptions, our FY 1983 program is consistent with the recommendation of the ACRS. This is due in large measure to the continuing dialogue which the Research Office has maintained with the Committee during development of the FY 1983 budget. This dialogue included Committee review of the FY 1982 version of the Long Range Research Plan, a technical review of the content of the FY 1982 program and the RES funding request submitted to the Commission prior to NRC's formal authorization request. It should be noted that as part of the technical review process the RES staff has, over the course of this year, conducted numerous briefings for the various topical subcommittees of the ACRS on specific elements of our programs. Examples of such elements are our programs on severe accident rulemaking, the rule on hydrogen control, the seismic margins, and waste management programs.

The differences in funding between the ACRS recommendations and our current requests reflect our balanced judgment of programmatic needs, the availability of resources to execute the programs and the need to provide for program continuity.

Question 46. Have changes been made in response to the NSOC Reactor Safety Research Review Group report in September 1981? Please enumerate them.

Answer.

The NSOC Reactor Safety Research Review Group Report contained forty-two recommendations which were considered by the Group to be particularly important. In many instances these recommendations related to increased emphasis in particular areas of research; in recognition of these, RES has intensified the efforts associated with such programs. As one example, the Review Group identified the need for a "major effort" on the development and evaluation of alternate shutdown heat removal systems for LWRs. At the time of the Review Group's deliberations, an investigation of these systems had been underway for several years as a follow-up effort to the "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants," NUREG-0438. In the past year, additional work on these systems has been initiated as part of NRC's program on Unresolved Safety Issues A45, "Shutdown Decay Heat Removal Requirements." Further, this concept is one of a broad spectrum of severe accident prevention or mitigation features being systematically evaluated for its risk reduction value and cost. This evaluation, which also includes the consideration of such features as filtered-vent containment systems, increased containment size, passive containment heat removal systems, etc., is being performed to provide one part of the technical basis for rulemaking and other regulatory activities related to severe accidents.

As another example, the second of the NSOC Reactor Safety Research Review Group's recommendations on existing research programs called for the program on understanding radioactive releases after severe accidents to include a broad and long range program on fission product release and transport. The effect of the NSOC recommendation was to strengthen the NRC's resolve to continue its research program in fission product release and transport and severe accident investigations. In early 1981 NRC summarized the current state-of-the-art in this area (i.e., fission product behavior) in a report, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents" (NUREG-0772); from this basis a broad program was launched to evaluate five risk dominant sequences, including degraded core and core melt accidents in four different plant types. These include a PWR large dry containment, an ince condenser containment, and two BWR containment designs (MARK I and MARK III). The findings of these evaluations will be confirmed with the on-going phenomenological research in this area. A broader program of research in all important aspects of the severe accident problem is underway and has been planned out through 1987, as described in the Severe Accident Research Program Plan (NUREG-0900) and the Long Range Research Plan (NUREG-0784).

Question 48. The ACRS also commented on April 5 that "the (Long Range Research Plan does not address research on LMFBRs or HTGRS to support commercial licensing, how many man-years of effort would be required? Does the NRC have contingency plans in the out-years of their program to address advanced reactor designs if such research becomes appropriate?

Answer.

If commercialization of a broad LMFBR program and supporting fuel cycle were to proceed there would be a need for both additional and more expanded programs. It has been estimated that the Office of Nuclear Regulatory Research would require 5 additional staff and a budget of $20-30 million per year. In addition, if GESMO proceedings were to be reopened, sigficant additional staff would be required in the Office of Nuclear Materials Safety and Safeguards and the Office of Executive Legal Director.

The NRC has prepared scoping plans for research in advanced reactors beyond CRBR. Such plans are of necessity rather general in nature because specific planning would need to be closely coordinated with any forthcoming national program for commercialization of advanced reactors to assure its relevance to the regulatory issues likely to arise with the specific program intended. If a more detailed plan had been developed in a "vacuum", it would require major revision before proceeding further when the actual program directions were defined. Because of this we do not believe it appropriate to proceed further with such plans at this time, and have instead focused our effort on dealing with Clinch River research issues, and addressing identified preapplication reviews issues for possible future HTGR commercialization (an effort in which we have obtained substantial input from the industry groups involved in HTGR program planning).

Question 47. The ACRS pointed out in an April 5 letter to Chairman Palladino concerning the Long Range Research Plan that "much still remains to be done in identifying . . . problems that represent the greatest potential contributors to risk." Is the NRC considering developing a systems approach for setting long-range research priorities which would respond to such critiques?

Answer.

Yes, NRC/RES is currently developing a plan by which results of risk assessment studies will be used to prioritize its programs. The basic aspects of this plan are: the periodic identification of those parts of the risk predictions which are important factors in the results and have significant uncertainty; the development of proposed methods for reducing these uncertainties; review of the technical merit, costs, etc., associated with various proposed methods; and the identification of the method to be pursued. This method will be used to prioritize research in those areas amenable to risk assessment. However, since certain areas of possible research are not amenable to such assessments, risk-important uncertainties will not and cannot be the sole determinant of NRC's research programs.

Question. Call in by Bouquard Staff

How much of the NRC research budget funds work at schools and universities, either directly or as subcontractors?

Answer.

The RES contracts and grants with universities for FY1982-83 is noted below:

(Dollars in millions)

	FY 82	FY 83
Contracts	3.7	3.4
Grants	0.5	0.7
	4.2	4.1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 8, 1982

The Honorable Marilyn L. Bouquard, Chairman
Subcommittee on Energy Research and
Production
Committee on Science and Technology
United States House of Representatives
Washington, D.C. 20515

Dear Madam Chairman:

Enclosed are the NRC staff responses to the twenty-seven post-hearing (May 18) questions received from your staff.

Sincerely,

John F. Suermann
for Carlton Kammerer, Director
Office of Congressional Affairs

Enclosure:
As stated

QUESTION 1. Could you tell me if there is any cost/benefit criterion applied to individual research programs and, if so, are such criteria published? How are decisions made as to the value of continued research in a particular area?

ANSWER.

There is no explicit cost/benefit criterion used for individual research programs. A variety of factors are considered in deciding on the value of research in a given area. They include the risk significance of the work, its regulatory significance, and its cost effectiveness.

a. Risk Significance

Ever since the completion of the Reactor Safety Study (WASH-1400), we have had available an overall assessment of risk contributors for PWR's and BWR's. Many improvements have been made in risk assessment and risk assessment techniques can be effectively applied in many areas; but we are also aware of the many technical areas, important to safety, in which risk assessments shed little light. A particularly important example of the latter is integrity of heavy section steel pressure barriers.

Lessons from risk assessment must be and are modified by judgment based on conventional engineering assessment in their use to judge the risk relevance of a candidate research program. The significance of a particular phenomenological uncertainty to the precision with which risks can be characterized is also important. As a criterion for doing research, the work proposed must significantly reduce or eliminate important uncertainty in an area significant to risk. In itself risk significance is of direct importance to regulatory decisions; the combination of risk significance with reducible uncertainty is of importance to research decisions.

b. Regulatory Significance

The regulatory significance of proposed research is a second important factor in setting priorities. The proposed work should be useful as a basis for current or expected regulatory decisions. Thus, research into a current licensing question of moderate importance on light-water reactors will be given priority over work on a major safety issue of advanced gas-cooled reactors because of its current regulatory significance. This aspect of priority is given special attention in the review of user needs and obtaining endorsement from the user offices.

c. Cost Effectiveness

The cost effectiveness of research programs is the third major factor in setting their priority. The obvious factors of cost and yield are foremost in judging a program, i.e., the results obtained per dollar spent. However, the research planner must also consider the timeliness of the research product and the fit with existing research capabilities. Reactor safety research purchased by the NRC is not an open market commodity; we are not buying part of a large body of activity. Rather the NRC is now conducting the largest part of the U.S. reactor safety research program. In that context it is very important to maintain the facilities and experienced researchers which are the core of the program. Of course, the NRC should not maintain facilities just because it is the major fund source for them. Rather, it is important that major shifts of allocated resources not be taken casually. A major capability not used for one year may cease to be a capability at all. The real people involved will move to real jobs elsewhere.

There is no neat equation for making these decisions on research priorities. Risk significance, regulatory significance, and cost effectiveness are debated again and again throughout the long-range planning process, the annual research budget review with the NRC user offices, the ACRS, the OMB and the Congress, in consolidated research planning with EPRI and other industry research programs and with foreign researchers, and finally even down into the continuing reassessment of current research activities as results are put into application by the regulators and the industry.

QUESTION 2. Is risk reduction potential explicitly taken into account when comparing the merits of research programs competing for the same resources? If not, will they be in the future? If so, when?

ANSWER.

Risk reduction potential has not, in the past, usually been taken into account explicitly in comparing research programs in broadly different technical areas, but is coming into use in planning within specific areas with some commonality, that is in technical areas which form a coherent set. The driving criterion for research is, of course, not risk reduction potential alone, but the degree of reduction of uncertainty per degree of risk significance. This is usually the basis for a user request for research; there is a need to reduce uncertainty in some area which is considered risk relevant. In one major area of NRC research, the Severe Accident Research Program (SARP), we are using explicit calculation of risk reduction potential in a systematic way. The primary purpose of the SARP is to determine what risk reducing changes to existing and future reactors are worthwhile. As this program generates the basis for regulatory decisions on design or operating features with risk reduction potential, it will also explicitly identify those areas of plant behavior where uncertainty and risk significance combine to warrant research. The SARP is underway now and its preliminary results will be made available by the end of this fiscal year. This program also includes our efforts to use risk significance extensively in our current planning of research on accidents involving damage to fuel or cladding.

QUESTION 3. I understand that you are cooperating with DOE as they develop a comprehensive reactor safety program. Do you expect to contribute to this program, once defined, in research areas in which NRC already has a strong program?

ANSWER.

Yes, we have already indicated to DOE in several meetings held in the spring and summer of FY 81 between the NRC Director of Research and officials of the DOE Office of the Assistant Secretary for Nuclear Energy that we would transfer research which seems more appropriate to them providing that DOE has been given the resources to carry out that work. We feel there is a appropriate role for the Department of Energy which could complement the NRC program. Examples of these functions include:

1. Assess whether NRC's research solutions to a specific nuclear safety problem are commercially feasible;
2. Maintain and operate large research facilities such as LOFT, that could have broader applications to energy research than the narrow nuclear-safety-oriented research objectives of NRC;
3. Provide a forum for dialogue among parties concerning research programs designed to address current operating reactor plant problems with broad implications; and
4. Assist industry in finding or developing solutions to safety issues, particularly for future plants.

QUESTION 4. The NRC research staff has proposed in the past that conservatisms in Appendix K of 10CFR50 be relaxed as a result of NRC-sponsored research. Your written testimony says that it "may be appropriate to simplify and improve Appendix K." Is research continuing to support a decision in this area? If so, when will work be completed? How much money is being spent on this effort?

ANSWER.

As a result of the research completed to date, the NRC staff could effect simplification in the Appendix K analyses required when plant design modifications are made. A broad revision of Appendix K will require completion of the ongoing loss of coolant and emergency core cooling research now scheduled for 1985. This program is being conducted with others at a number of large multipurpose research facilities with a total cost of operations through 1985 amounting to $50M. In the interim, consideration will be given to minor modifications of Appendix K requirements in the licensing process prior to the rulemaking.

QUESTION 5. Is one of the goals of SSMRP to determine the degree of conservatisms in current NRC licensing criteria for seismic design? If so, when do you expect this goal to be achieved?

ANSWER.

Yes, determining the degree of (or lack of) conservatism of current licensing criteria is an objective of the SSMRP. The SSMRP methodology will be used to evaluate regulatory practices starting in FY 84 with completion in FY 86. This assumes that the methodology will be validated in FY 84. Preliminary estimates of conservatisms will be available in FY 84.

QUESTION 6. The ACRS said about this program "application of the current methodology to a given plant requires an inordinate amount of time and money and a knowledge about the plant that may be difficult or impossible to obtain." Would you respond to this comment and describe the data which must be supplied about a particular plant in order to subject it to these codes?

ANSWER.

It is estimated that it would require a $1.5 million effort to analyze another nuclear power plant to the extent and detail that Zion is being analyzed as part of SSMRP. The results of the Zion study and sensitivity studies being performed are expected to simplify seismic response modelling and event/fault trees resulting in simplified methodology for performing seismic risk analyses. It is expected that the simplified methodology will reduce considerably the estimated cost identified above for Zion.

The Seismic Hazard Characterization Project will produce seismic input data that can be applied to any site east of the Rocky Mountains. Therefore, data needed to analyze a nuclear plant will consist of site soil information, building and subsystem design data (and seismic analyses, if available), and systems design information.

Data needed for seismic risk analyses is contained in licensing documents (FSAR, Tech. Specs, Operation Manual, etc.) which are in the NRC's possession, plus design information (drawings, models, etc.) which are normally available only through the plant's architectural engineering (A/E) firm and reactor supplier.

QUESTION 7. What was the cost of supplying input data for the Zion plant? Do they have to be refined to confidently assess the seismic safety margins in the Zion plant? For how many other plants do these input data currently exist?

ANSWER.

In the Zion analysis, the A/E charged LLNL for their effort in supplying the information needed. The cost was roughly $100,000 (or less). The development of generic/simplified SSMRP methodology should reduce the amount of input data, but A/E (and utility) cooperation would still be needed.

The Zion plant was used to develop a complex seismic risk methodology. This elaborate methodology will be simplified for applications on other plants. Particular, plant specific, input data will be necessary, and it will need some refinement. However, this refinement will be at a much simpler level than done for the Zion plant. Because we do not expect to perform a seismic risk analysis using the complex methodology developed for Zion, there will be no need to develop that type of input data for evaluating other plants.

QUESTION 8. How important are modeling uncertainties to the results of the seismic analysis done in SSMRP? Will additional research on modeling be needed before conservatisms in the NRC regulations can confidently be determined?

ANSWER.

Modeling uncertainties develop from several sources including inaccurate information on drawings and our ability to introduce into the analytical process, usually a computer program, a completely true representation of all the details that affect component behavior. Preliminary SSMRP results have shown that, for some areas such as piping response, the contribution of modeling uncertainties has a significant effect on results. Development of simplified models, as addressed in Question 6, should reduce the number of parameters involved in the analysis and the need for research to validate these parameters.

Additional research to evaluate and reduce uncertainties due to modeling errors will be done selectively based on (1) results of sensitivity studies to develop the simplified models and (2) actual seismic risk calculations. Efforts will concentrate on areas where the results would yield a significant reduction in uncertainties important to risk or reduction of an onerous burden on licensees. Research on piping system modeling is underway which should contribute in both these areas. It is now apparent that present NRC requirements for piping system design result in complex models and very rigid piping systems. Research to improve our understanding of structural behavior may show that, in some cases, requirements imposed by NRC with the intent of increasing conservatism in piping systems may have resulted in actually increasing risk in the nuclear power plant as a whole.

QUESTION 9. Your testimony addresses your objective of developing a technical basis for sound decision making. Can you explain how your research on the behavior of degraded fuel, for example, might enter the Commission's decision process?

ANSWER.

The Commission's decision process on regulatory treatment of accidents involving damage to fuel and release of fission products will depend, in part, on our understanding of degraded core behavior since the core is the source of the radiological hazard and of the loads on the containment. Current analytical methodologies used in Probabilistic Risk Assessments (PRA's) contain many assumptions on the phenomenology of accident sequences and degraded cores which, depending on the assumptions employed, either increase or decrease the risk for a given accident scenario. Such uncertainties affect the estimates of hydrogen evolution, fission product source term and timing, melt progression and containment loading, and melt coolability. The damaged fuel research program is designed to reduce these uncertainties via a multi-faceted programmatic approach based upon integral in-pile tests, separate effects experiments, analytical model development, and the examination of the TMI core. The goal is to produce a PRA methodology based on experimentally validated codes to be used for decision-making on risk and cost/benefit attributes of possible design and operational improvements on specific plants.

QUESTION 10.	Your program to examine containment integrity evidently includes numerous overpressure tests of scale model containment structures ranging in size from 1/32 to 1/10 of typical containments for operating reactors. What, if any, technical difficulties might have to be overcome to apply the results to full-scale containment structure?

ANSWER.

The objective of the containment integrity program is to develop a technology to reliably predict certain performances of the containments loaded well beyond design levels. The tests on scale model containment structures are not intended to be replica models of existing containments. They are designed to provide a data base against which predictive calculations can be checked. Existing computer codes have the potential to make these predictions but have not been verified for application to this problem.

The significant technical problems associated with the application of the results to full-scale containments are:

1. It can be difficult to design scale test models that adequately simulate the appropriate test parameters associated with the containment design and details of construction techniques. The smaller the scale of the test model the more difficult this can be.

2. If the containment capacity is limited by leakage around penetrations, rather than by gross failure, the simulation of seal tolerances and seal material degradation with time is very difficult in small scale models. An independent research project is now being formulated to study penetration leakage on larger detailed models.

QUESTION 11. What were the conclusions of your study regarding the use of LOFT for investigating the desirability of requiring that the main coolant pumps be turned off or left on during a small-break LOCA? Is there any danger that prescriptive recommendations from such a study might lead to erroneous results during some accident scenarios?

ANSWER.

LOFT studied the effects of primary coolant pump operation during two small-break LOCA tests, L3-5 and L3-6. The results of those tests led to the conclusions that running of the primary coolant pumps resulted in lower fuel cladding temperatures. However, the "pumps-on" case also resulted in higher reactor coolant loss through the break, thus, when the pumps were later turned off there remained insufficient coolant to cover the reactor core. This resulted in a fuel temperature excursion. Thus the NRC staff recommendation of pumps off after indications of a small break appears to be correct.

Further testing of LOFT has resulted in generation of a pump electrical current demand versus time curve which can be used by reactor operators to determine, uniquely, the accident mode the plant is experiencing. Specifically, small-break LOCA's, heatup transients, and cooldown transients each have a unique pump current "signature." This information has been provided to NRR and the industry to assist in developing an unambiguous pump trip criteria. This should allow a non-prescriptive recommendation to be worked out.

The current pumps-off recommendation has been modified for the so-called 2 x 4 (two loop four pumps) plants so that the plant operator may leave one pump per loop running so long as the other pump was operable when it was shut off.

Question 12. How many codes have been developed under NRC sponsorship? Would you provide a list of these codes for the record and describe the purpose of each.

ANSWER

Many codes have been developed under NRC sponsorship, a good number of which were low budget special purpose codes developed for short term use. A list of the codes now in use or currently nearing completion which NRC has developed is attached.

Code Title	Code Purpose or Use	Regulatory Application
TRAC-PD2	Detailed (3-Dimensional) best estimate calculations for large break LOCA in PWR's	• Audit vendors PWR calculations. • Evaluate safety margins in PWR design and base accident analysis. • Provide PWR analysis base for rulemaking procedures.
TRAC-BD1	Same as above, except for BWR's. Also, limited capability to address small break LOCA and transients in BWR's.	Same as above, except for BWR's.
TRAC-PF1	Fast running version of TRAC for calculations of small break LOCA's and operational transients in PWR	• Assessment of plant response to multiple (compound) failures and operator actions. • Analyzes in support of rulemaking procedures and for evaluation of margin of safety.
RELAP 4	1-Dimensional thermal-hydraulic calculations of LWR LOCA	• Best estimate and EM type calculations of LOCA behavior.
RELAP 5	1-Dimensional thermal-hydraulic analysis of LWR and experimental facilities (including LOCA)	• Evaluation of licensee submittals and independent assessment of LWR transient behavior.
COBRA/TRAC	Detailed (3-Dimensional) best estimate analyses of LOCA consequences in PWR plants featuring Upper Head Injection of emergency coolant	• Same as for TRAC-PD2, except for its focus on plants featuring Upper Head and/or Upper Plenum injection of emergency coolant.
COBRA-TF	Detailed thermal-hydraulic analysis of individual subchannels in LWR's	• Assess peaking factors. • Evaluate consequences of flow blockage induced by clad swelling.
RAMONA-3B	Best estimate analyses of anticipated transients without scram (ATWS) in BWR plants. Considers 3-Dimensional neutron kinetics, with all feedbacks.	• Assess consequences of ATWS, with either symmetric or asymmetric control rod arrangement. • Audit vendor analyses. • Provide analyses in support of rulemaking.

Code Title	Code Purpose or Use	Regulatory Application
FRAP-T	Detailed fuel behavior	°Audit vendors calculations
FRAPCON	Detailed fuel behavior code for stored energy computation and an initializer for FRAP-T	°Audit vendor calculations and codes
SCDAP	Detailed fuel behavior code for analyzing degraded core behavior	°None as yet. Will be used to benchmark and assess PRA (risk) codes
FASTGRASS & GRASS-SST	First principal codes describing fission product release from fuel	°Audit vendor calculations
TRAP-MELT	Assess fission product tranport and attenuation within LWR reactor coolant system during severe accidents	°Used in reassessing severe accident radiological source terms
COMMIX-1A	Three-dimensional transient non-equilibrium thermal hydraulics single phase with porous medium capabilities	°Use LMFBR safety issues - natural circulation temperature gradients, etc.
K-Fix (3D, FLX)	Three-dimensional fluid-structure interaction	°Dynamic loads on core barrel after a large-break LOCA
SOLA-3D	Three-dimensional flows with turbulent mixing and minimal numerical diffusion	°Fluid mixing in the cold-leg and downcomer under pressurized thermal shock conditions
WRAP-EM	Conservative one-dimensional calculations for a large-break LOCA	°Audit of vendor evaluation model calculations

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Code Title	Code Purpose or Use	Regulatory Application
SSC-L	Complete systems code analysis of loop-type LMFBR plants	Assess effects of pipe breaks on core temperatures in CRBR Assess core coolability by natural connection in CRBR
SSC-P	Complete systems code analysis of pot type LMFBR plants	Same as SSC-L except for pot-type plants (same versions of CDS)
BRENDA	Fast running systems code analysis of loop-type LMFBR plants	Provide input to accident delineation and risk assessment studies
EPIC	Models ejection of molten fuel into coolant during LOF-TOP accident	Assessment of reactor core response to serious accidents
CONTAIN	Models a reactor containment system and predicts loads imposed by accidents	Assessment of the adequacy of structural design of reactor containment systems
CORCON	Models the interaction between molten reactor-core materials and concrete during fuel melt accidents	Assess threat to containment from concrete ablation
LAFM	Models reactor fuel behavior under transient overpower conditions	Assessment of fuel disruption and relocation during an overpower accident
SAA	Version 1 designed for NRC to assess the vulnerability of nuclear fuel cycle facilities to diversion.	It is intended for use by material control and material accounting license reviews to test compliance with the physical protection and material accounting upgrade rules. It uses Brolian path-finding routines.
SVAP	Designed as a method of analyzing the safeguard systems at nuclear facilities for vulnerabilities.	SVAP addresses on class of threat: theft or diversion of nuclear materials by nonviolent insiders, acting individually or in collusion. It uses Brolian path-finding routines.
ASM	An adversary scenario interference technique designed to integrate various forms of safeguards information.	Provides an evaluation and ranking of complex safeguards alternatives to aid nuclear safeguards decision making.
BEIRMOD	To predict the health effects from low level radiation.	Used by ORPB, DFO to estimate health effects for various conditions of exposures.

Code Title	Code Purpose or Use	Regulatory Application
AEGIS	Simulator to test fish impingement models.	Being developed for NRR Environmental Systems Branch for hearing preparations and EIS's.
CONCEPT	Capital cost estimator for NPP, and coal fired alternative steam supply systems developed by United Engineers.	Used for cost analyses by support group at Oak Ridge National Lab.
STRIPE	A computer code for simulating the striped bass population in the Hudson River.	Developed for NRR for use in Indian Point hearings and for future EIS's, by ORNL.
CONCEPT/OMCOST	A code to estimate the capital and non-fuel operating costs of nuclear power plants.	Used in the licensing process to evaluate costs of nuclear vs. alternative electrical generating facilities.
INREM II	Calculates the internal radiation dose equivalent to human organs resulting from the intake of a radionuclide by inhalation or ingestion.	Used by Radiological Assessments Branch (NRR) in licensing reactors.
SPAHR	Calculates morbidity and mortality in populations exposed to effluents from both coal and nuclear fuel cycles.	Used by Radiological Assessments Branch (NRR) in licensing reactors.
DOSFACTER	Calculates the dose to human organs resulting from external exposure to photon and electrons.	Used by Radiological Assessments Branch in licensing reactors (NRR).
SWIFT	Three dimensional finite difference simulation of radionuclide transport from HLW through groundwater to the biosphere, accoundting for effects of heat transport (HLW is a heat source) and salt solution. Both near field (near the HLW) and far field (far from the HLW) effects can be simulated.	Evaluation of HLW disposal license applications from DOE.
NWFT	Simplified calculation of radionuclide transport from HLW, ignoring thermal effects. Allows decay chains of up to only three radionuclides with equal retardation. Applicable only to far field situations.	Same as SWIFT. Used especially in risk and sensitivity analyses when the range of controlling parameters is large and uncertain.
NWFT/DVM	An extension of NWFT to allow for any number of radionudlides to be considered in a decay chain and to allow for unequal retardation among the radionuclides.	Same as NWFT.

Code Title	Code Purpose or Use	Regulatory Application
DNET	Simplified near field simulation of the HLW's host material due to thermal effects, salt dissolution, salt creep, and mechanical loading.	Evaluation of performance of host material containing emplaced HLW.
LHC	Uses Latin Hypercube Sampling, an economical statistical parameter selection method for choosing families of parameters in conducting sensitivity analyses to determine the influence of independent variables on dependent variables in large complex calculations.	Selection of values from known distributions of parameters for use in SWIFT, DNET, NWFT, and NWFT/DVM.
STPDELT	Uses statistical regression methods to identify and rank importance of those independent variables which control a given set of data.	Identification of the significant input parameters which control the outputs of SWIFT, NWFT, NWFT/DVM, and DNET.
PATH1	Implements environmental transport model and transport to man model to simulate the transport of radionuclides from a given source to man.	Predictions of radionuclides released to the biosphere by SWIFT, NWFT, and NWFT/DVM provide the numerical values describing source of radionuclides for PATH1.
HEALTH	Applies dose conversion models to radionuclides adsorbed, inhaled, or ingested by man to assess the damage to individual organs caused by the radionuclides.	The radionuclide decay chains used in HEALTH are those taken from the outputs of PATH1.
SIM	Soil-structure interaction using half-space solution.	As a simplified analytical check on licensee calculations.
SLAM	Soil-structure interaction code using finite elements.	Not used by NRR:SEB.
SLAVE	Analyzes deconvolution of earthquake motion.	Not used by NRR:SEB.
SMACS	Code to estimate structural and component seismic responses and uncertainties.	Available in 1982 for independent calculations.
SEISIM	Code to estimate probability of failure of structures and components and seismic risk.	Available in 1982 for independent calculations.
HAZARD	Code to assess the seismic hazard and develop and correlate information to generate time histories.	Used by NRR in SEP to develop time histories.

Code Title	Code Purpose or Use	Regulatory Application
WIPS	Structural Finite Element code used to evaluate pipe whip and impact.	Available to NRR by February 1982 for independent Calc.
PRAISE	Probabilistic crack growth computed for piping systems.	Used already for asymmetric blowdown calculations in PWRs. Extended to BWRs in '82 by including stress-corrosion cracking.
PSAFE 1 & 2	Estimates piping response to dynamic environments.	Used to evaluate seven codes for twelve plants in NRR safety assessments.
PARET	Extracts eigenparameters from dynamics tests.	Basically used only for research.
Nozflaw	Calculates fracture mechanics stress intensity values along crack front - Just released to general public.	Can be used by NRR and utilities to evaluate flaws found in BWR and PWR nozzle corners to develop requirements for repair or continued operation.
OCA-1	Fracture mechanics code for predicting initiation and propagation of cracks in PWR RPVs under LOCA and pressurized Thermal Shock scenarios.	First version of code complete and put up for use by NRR personnel.
Whips	Large deflection, nonlinear, elastic-plastic pipe whip code for analyzing the deflection of piping subsequent to postulated guillotine, double-ended pipe breaks. Can analyze long runs of piping, including nonlinear pipe restraints and snubbers and pipe-to-pipe and pipe-to-wall impact.	To be used to develop data for NRR for the Revision of Reg. Guide 1.46. Code to be complete by end of December 1981.
OCTAVIA	Probabilistic linear elastic fracture mechanics code for analysis of pressure vessels subjected to overpressure transients when pressure vessel is uniformly at any given temperature.	Used by NRR to address the overpressure of RPV problem.
MULT-NOZZLE	Program for the analyses of closely spaced nozzles in pressure vessels.	Was used to develop data for the Revision of ASME Boiler and Pressure Vessel Code, Section III.
FLANGE	Code developed for the analysis of bolted flange connections with ring type gaskets.	Was used to develop data for the Revision of ASME Boiler and Pressure Vessel Codes, Section's VIII and III.
WELDS	Program to calculate the residual stress fields around circumferential girth welds in piping. Data developed to bound stress fields in piping of BWRs subject to intergranular stress corrosion.	Program picked up and used in EPRI studies on pipe cracking.

Code Title	Code Purpose or Use	Regulatory Application
COPPER	Mathematical model for predicting the transport of copper released from nuclear station to marine environment	°Used in assessing copper source term impacts
CHLOR	Kinetic model for predicting the composition of chlorinated water discharged from nuclear station using fresh water cooling systems	°Used in assessing chlorine source temperature impacts
CIRCU	Two-dimensional model to predict wave- and wind-induced nearshore circulation patterns in lakes and coastal areas	°None as yet. Model recently validated. Will be used for new sites on lakes and coastal areas. Expected to be used on sites located on lakes and coastal areas where beach erosion is indicated to be a serious problem
CIRC SED	Two-dimensional model to identify regions of the nearshore circulation zone where sediment erosion and accretion are likely.	°None as yet. Model validation results being reviewed. Expected to be used on sites located on lakes and coastal areas where erosion is indicated to be a serious problem.

QUESTION 13. What additional research activities would be necessary for the NRC to process an application to operate the Barnwell Nuclear Fuel Plant?

ANSWER.

At this time, no additional research activity has been identified as necessary for NRC to process an application to operate the Barnwell Nuclear Fuel Plant.

QUESTION 14. Has NRC conducted any research defining the impact on safety of mandating a high point vent as a result of concerns raised by TMI? Has the safety improvement credited to this requirement been compared to safety degradation due to the additional penetration required? If so, when will these results be available?

ANSWER.

NRC has not conducted research to quantify the impact on safety of adding the TMI-inspired high point vent. In general, this change does not lend itself to easy risk quantification. The safety decrement is a probably small increase in small LOCA probability, an initiating event already dominated by other contributors such as reactor coolant pump seal failure. The safety increment is an unknown increase in the ability of the operator to cope with transient and small LOCA sequences where vapor pockets have formed which hinder cooling flow. On balance it appears that the safety increment is larger than the safety decrement, but it has not been quantified.

QUESTION 15. Is NRC conducting research to examine proposed reactor vessel water-level indicators? Is the reliability of readings from these instruments during a variety of accident scenarios being considered for possible adverse effects on safety?

ANSWER.

NRC/RES is continuing to conduct research to examine several proposed reactor vessel level instruments and to assess their potential effects on safety. Specific research programs and their status are as follows:

> ORNL is drafting the final topical report on the evaluation of the Westinghouse Pressure Differential System tested at the semi-scale facility under various small-break LOCA and natural circulation conditions.
>
> ORNL is also drafting the final topical report on a completed evaluation of the heated thermocouple technique for water level measurement.
>
> ORNL is evaluating the torsional ultrasonic technique. Results to date are very promising. Program completion is planned in FY 83.
>
> Under an NRC grant, Penn State University is studying the feasibility of using an ex-vessel neutron detector for reactor vessel water level measurement.

The main objective of testing the prototypes in our LOCA facilities is to evaluate the performance of instruments and the reliability of the reading during a variety of accident conditions. The limitations of the instruments are also being assessed to minimize possible adverse effects on safety due to erroneous readings.

QUESTION 16. When addressing human engineering in your Long Range Research Plan you use the word "coordination" to describe "exchanges of program plans and research results." Would you qualify your meaning of the word as used in your testimony, particularly in the context of programs which you cite as coordinated with EPRI and INPO?

ANSWER.

On Page 7-4, para. 7.1.3, of the plan, Relationship to Other Programs, we stated, "NRC's human engineering research is coordinated with the activities of other organizations (Table 1) to ensure complete coverage of important topics without unneccessary duplication. Coordination occurs most often through exchanges of program plans and research results and participation in working groups by NRC personnel." The specific NRC research tasks which EPRI and INPO are informed about and with which close liaison and communications are maintained are task analyses of nuclear power plant operating crews, maintenance personnel and management personnel functions. Other research activities are programs in human performance data collection using simulators, maintenance modeling, development of display design criteria, and evaluation of various display formats. All NRC research reports are transmitted to EPRI and INPO. Personnel from both organizations are now and will be invited participants in NRC research review groups which evaluate on-going human factors research programs. Frequent contact between NRC and INPO/EPRI staffs are maintained and routine exchanges of information at meetings occur. The human factors staffs of the respective organizations also meet to describe the current and future research programs. As a consequence of this program data exchange, unnecessary duplication of ongoing and future activities is avoided. A positive effort to avoid unnecessary duplicative research as well as to encompass critical research needs are major objectives of NRC, EPRI and INPO. Both EPRI and INPO keep the NRC human factors staff informed of their research tasks and products by scheduled information exchange activities, transmission of research reports, and staff interaction among the organizations. In conclusion, the word, "coordination", is meant to describe all of the cooperation and interchange of information that is ongoing among NRC, EPRI and INPO.

QUESTION 17. I understand that you are awarding a $5 million contract to Battelle Columbus for the study of high-level waste packages. How does that study differ from current investigations underway at Battelle under DOE sponsorship?

ANSWER.

The NRC-sponsored study at the Battelle Columbus Laboratories (BCL) will investigate the physical and chemical processes that degrade the ability of high-level waste packages to contain the waste. The NRC research at BCL is focused on developing quantitative models that can be used to identify and evaluate the factors that affect long-term behavior of waste packages, which provide the source terms of radionuclides available for groundwater transport. The objective of the NRC research at BCL is to improve the NRC understanding of those factors in order to assess uncertainties in DOE's performance assessments submitted for NRC licensing action.

The DOE research in this area is developmental in nature, and is focused on evaluation of relative performance of various candidate materials and engineered designs based on relatively short-term standard tests. Although DOE's program may develop similar quantitative models, their primary emphasis will be on applicability of the models using current information and data to develop waste package design. Since the assessment of long-term behavior of waste packages and radionuclide source terms are very critical in making licensing decisions, it is necessary for NRC to be able, independently, to assess the completeness and validity of the DOE predictions of long-term waste isolation performance.

The BCL's study will be closely coordinated with DOE's related program, which includes studies by Material Characterization Center at Battelle Pacific Northwest Laboratories. This will allow us to take full advantage of DOE research results and to ensure that the BCL work is aimed at relevant licensing decisions.

The Office of Nuclear Waste Isolation (ONWI) at Battelle, Columbus is established to be managed completely independent from the BCL. The BCL does not conduct any DOE-sponsored research on this subject and will not undertake any such work during the conduct of the NRC contract.

QUESTION 18. The Subcommittee understands that some NRC research is also being conducted outside the Division of Nuclear Regulatory Research in the Divsion of Waste Management. The Subcommittee understands that the NRC is conducting a $30 million program with 105 people in the Division of Waste Management. Please describe this program, state whether it is duplicative with DOE research and describe any other areas where any NRC division outside of the Research Organization is conducting or contracting for research.

ANSWER:

The Division of Waste Management has 105 people for FY 1982: 29 involved with low-level waste management (LLW) activities; 34 with uranium mill tailings (UR); and 42 with high-level waste management (HLW). The overall budget for the Division of Waste Management for FY 1982 is approximately $11.8 million; $2.3 million allocated to LLW activities; $1.6 million to UR: and $7.9 million to HLW. In contrast, the magnitude of the DOE HLW program is currently at about $200 million per year.

Work being performed by NRC in the area of HLW is fundamentally different from that of DOE for two reasons:

First, DOE, under the Energy Reorganization Act of 1974, will be a license applicant to NRC for disposing of HLW in deep geologic repositories. In this case DOE is more like utilities or other licensees than like another federal agency. As a result, NRC will utilize DOE research to the extent practicable but will also undertake duplicative work, where deemed necessary, in order to assure that it fulfills its statutory responsibilities. For example, radionuclide transport modeling is still very much a developmental area, and could potentially be contentious during the licensing proceedings. Therefore, NRC is undertaking independent confirmatory research and development work on these transport models in order to be in a position to independently assess DOE's modeling assumptions and activities at the time of license submittal as well as providing guidance to DOE in advance of receipt of an application.

QUESTION 19. How much is currently being spent on waste package R&D?

ANSWER.

The FY 1982 budgets for research regarding High-Level and Low-Level waste packages are $1.4 million and $1.1 million, respectively. The High-Level Waste package program includes studies of long-term performance of waste form, canister/overpack, and backfill materials that the DOE is considering. The Low-Level Waste package program is focused on evaluating and establishing test procedures to provide guidance to licensees for confirming the characteristics of waste packages and for predicting the long-term performance of various waste forms that utilities and industries are using or plan to use.

Second, DOE's job is one of developing a product. DOE is now developing the production details of waste forms and is beginning work to develop specific sites into working repositories.

NRC programs, on the other hand, are aimed at assuring that all of the important safety issues are identified and are being adequately addressed by DOE's program. NRC's program is directed to understanding the fundamental questions and uncertainties associated with long-term isolation of HLW rather than product or site development. We also want to assure there are no gaps or flaws in the DOE program. We want to flush out the problems early while there is still time to resolve them without being on the critical path.

In the HLW area, the NRC has structured a regulatory process which is intended to assure that important issues are raised and resolved early and not taken up at the time when they can be disruptive and lead to delay. Given that a HLW repository is a unique, first-of-a-kind facility, and given the long lead times required to develop designs and resolve basic questions of site suitability, such extensive prelicensing interaction between NRC and DOE is essential. The purpose of NRC work is to make certain that there is no question about what will be needed for licensing before DOE makes major commitments in their program and to otherwise be prepared for a rapid licensing review. Waiting to address questions about the direction DOE is taking in its program until the formal licensing process begins is the sure way to a protracted, highly contentious licensing proceeding.

Regarding the question of where any NRC division outside of the Research Organization is conducting or contracting research, the total NRC WM program is coordinated through the Division of Waste Management which provides overall Program Area Management for the NRC Waste Management programs. The Division of Waste Management integrates the activities of all NRC offices providing support to the Waste Management programs under a charter established by the Executive Director for Operations (EDO). As part of the program area management, all waste management contracts are coordinated and integrated in the formulation phase. Further, all NRC Technical Assistance and Research contracts are then reviewed by the interoffice NRC Waste Management Review Group to check for unnecessary duplication, overlap, and program relevance. This process assures that no waste management research is conducted outside the Office of Research.

We have coordinated our program very carefully with the DOE and other federal agencies, such as the EPA and USGS, to assure there is no unnecessary duplication of efforts. For example, we provide DOE with descriptive summaries of all of our contracting actions. We have in some instances eliminated from our program plans research that could be more appropriately performed by the DOE and thus significantly reduced project budgets over the past several years.

Regarding the question of where any NRC division outside of the Research Organization is conducting or contracting research, the Office of Nuclear Materials Safety and Safeguards, the Office of Nuclear Reactor Regulation and other NRC offices contract for technical assistance to their licensing or regulatory functions. Technical assistance provides supplemental support of technical capabilities on specific regulatory issues requiring rapid resolution.

QUESTION 20. How much is currently being spent on waste form R&D?

ANSWER.

$0.5 million of the FY-1982 waste management budget is allocated to research on high-level waste forms. The main portion of this research is devoted to the study of borosilicate glass.

The FY 1982 budget for low-level waste form is $0.5 million. Such waste forms as hydraulic cement, vinyl ester styrene, and bitumen are being studied to determine their limits of applicability for various waste streams including normal nuclear reactor wastes, incineration ash, and decontamination wastes.

QUESTION 21. What is the relationship between NRC's research on alternative waste forms and related DOE research?

ANSWER.

The objective of the NRC research is to improve understanding of factors affecting long-term performance and to answer questions relating to licensing decisions. At present, it appears that the main thrust of the DOE effort is toward borosilicate glasses as a waste form; consequently most of NRC's resources for research on waste forms is devoted to borosilicate glass. A small amount of effort has been devoted to an alternative waste form, supercalcine, and additional research is planned to assess synthetic mineral waste forms. Should DOE change emphasis of its program, this would be reflected in a corresponding change in the NRC programs to permit answers to be developed to questions whose importance was increased by the DOE action.

QUESTION 22. This February, the NRC published the Proposed Policy Statement on Safety Goals for Nuclear Power Plants. What is the present status of the proposed policy? Has an implementation plan been developed, and how would it affect safety research? When available, please provide us with a copy.

ANSWER.

The public comment period for this proposed policy statement has just closed. A series of four public meetings around the country were also held during the comment period to elicit further comment. The Commission has assigned the task of developing an Implementation Plan to the Office of Nuclear Reactor Regulation. Since it would also affect research, we are working with them to prepare the Plan. The Policy Statement on Safety Goals will affect research work in that risk assessments are being used more and more to evaluate research priorities; the existence of even a trial quantitative safety policy will be useful in assessing those priorities. In addition, there will be some continuing research work directly related to final development of the quantitative safety policy. The Plan should be complete soon and will be sent to the Commission. There is strong sentiment from the extensive public comments to reissue the policy statement for comment along with the Plan. It is not clear at this time what course the Commission will follow.

QUESTION 23. Has NRC decided on the allocation of the entire 10.5 million in R&D funds for LMFBR?

ANSWER.

The NRC has decided substantially on the allocation of the 10.5 million in R&D funds for LMFBR research in FY 1983. The scope and objectives of a proposed probablistic risk assessment for the CRBR plant have not been resolved. The risk assessment, which amounts to $1.2 M, is the only major outstanding item.

QUESTION 24. Apparently, the NRC research program on uranium mill tailings is focused on covers to attenuate radon emissions. However, a recent memo to Leonard Bickwit (General Counsel of NRC) from NRC staff stated that the effects of radon from tailings are judged by the staff and others to be of little or no real significance and by themselves would not justify uranium mill tailings regulation. The DOE has also stated that the focus on mill tailings regulations should be to (1) control erosion from piles, (2) prevent misuse of the tailings, and (3) prevent intrusion into the tailings, not particularly attenuation of radon emissions. Why then is the NRC spending $2.6 million in FY 1982 and another $3.2 million in FY 1983 to do this research? Why is this type of research not done more appropriately by DOE or industry?

ANSWER.

The NRC's proposed FY 1983 uranium recovery research program includes evaluations and assessment relating to the following issues: long-term and interim stabilization of tailings piles; tailings dewatering and neutralization techniques and other alternatives for immobilizing toxic materials; disposal techniques allowing contact with groundwater (i.e., deep mine disposal techniques); effluent and environmental monitoring methods and equipment; methods for minimizing groundwater contamination from in situ mining; bottom liner and sand/slime segregation alternatives; assessment of leachate movement from ponded tailings; methods for predicting effects of long-term geomorphic processes; application of geophysical methods for site characteristics; and radon attenuation properties of cover materials. The research on the properties of covers for attenuating radon emanation from uranium tailings piles is needed to provide the NRC staff with reliable models for evaluating proposed radon attenuation covers and appropriate procedures for verifying the effectiveness of inplace covers prior to and after license termination. The parameters of most concern include moisture, particle size, and compaction. Since this information is specifically needed for decisions in NRC's licensing program, it appears appropriate that the research be sponsored by the NRC rather than DOE or industry. The funding of $2.6 million in FY 1982 and $3.2 million in FY 1983 are for the total uranium recovery research program in FY 1982 and FY 1983, respectively. Funding for the radon attenuation project is only a fraction of the overall uranium recovery program. The funding for this task is $250,000 in FY 1982 and the proposed funding for FY 1983 is $200,000. The research project on the attenuation of radon is scheduled to be completed in FY 1983.

The above-referenced memorandum to the General Counsel was sent by three members of the NRC staff and does not represent the NRC's official position on the significance of the health effects that could result from radon emitted from uncovered piles of uranium mill tailings. The NRC's evaluation of those potential health effects are contained in the Generic Environmental Impact Statement (GEIS) which supports the Commission's Uranium Mill Regulations.

This GEIS makes it clear that radon exhalation is but one of the several sources of radiation exposure arising from mill tailings. The research program reflects the regulatory interest in erosion, leaching effects on groundwater, intrusion and radon as factors affecting public health and safety and requiring consideration in the licensing process.

QUESTION 25. NRC's research program includes research on transuranics, including developing a methodology for determining the concentration of transuranics such as power reactor wastes. Would this same methodology be effective with other forms of transuranics such as those produced by heart pacemaker manufacturers or decommissioning commercial nuclear fuel fabrications facilities?

ANSWER.

The methodology referred to in the question uses "indicator" radionuclides, such as Co-60, in power reactor wastes to estimate the concentrations of other radionuclides such as transuranics. Since these indicator radionuclides are not present in nuclear fuel or pacemaker batteries, the methodology is not readily transferable to these wastes.

QUESTION 26. Is the NRC transuranic research program designed to address the issue of allowing for intermediate level radioactive waste disposal facilities (perhaps for the 10 to 100 nanocuries per gram range) such as the French and the Germans are developing? If not, what is the purpose of this research?

ANSWER.

The ongoing NRC waste management program does not specifically address the issue of disposing of transuranic wastes in the range of 10-100 nanocuries. However, the information regarding geochemical and radiochemical characteristics of Pu, U, Am, and Pa is generic in nature and will be applicable for assessing various proposals for the disposal of transuranic wastes in a wide range of concentrations.

QUESTION 27. Have the NRC studies of power plant sitings included an analysis of siting aspects for HTGRs? Has there been any research on the comparative risks and benefits of HTGRs, relative to LWRs, such as generic accidents and potential release?

ANSWER.

We are performing extensive LWR siting analysis using LWR Siting Source Terms which are source terms for sequences which are representative of different classes of severe accidents. For HTGRs, we are doing a similar study in which we will quantitatively evaluate accident sequences involving the loss of primary safety functions (e.g., decay heat removal) taking into account their nature and timing, in addition to qualitatively reviewing the resistance of HTGR designs to external events such as earthquakes and fires. The principle objective of this study is to define the significant HTGR risks so that a basis exists to judge the risks to the general public taking into account the specific features of HTGRs relevant to siting. The objective in these efforts is to deal with HTGRs as HTGRs with their special characteristics and not try to deal with them only by comparison to LWRs.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 11, 1982

Honorable Marilyn L. Bouquard, Chairman
Subcommittee on Energy Research and Production
Committee on Science and Technology
U.S. House of Representatives
Washington, DC 20515

Dear Madam Chairman:

Attached are ACRS responses to questions received from your staff. These supplement the testimony of Dr. Chester P. Siess before your Subcommittee on May 18, 1982.

Sincerely,

Raymond F Fraley

Raymond F. Fraley
Executive Director

Question 1

Would you explain the basis for ACRS' contention that the distinction between confirmatory research and research to improve reactor safety is no longer useful?

Response

In its report to the Congress on the FY 1981 NRC Safety Research Program (NUREG-0657), the ACRS stated the following with regard to "Confirmatory Research" vs. "Research to Improve Reactor Safety":

> The Energy Reorganization Act of 1974, which effected the separation of the Atomic Energy Commission (AEC) into the separate entities of the NRC and Energy Research and Development Administration (ERDA), established within the NRC an Office of Nuclear Regulatory Research (RES). In view of the legislative history (Conference Report), the role of RES was accepted by the NRC as being limited to "confirmatory research," presumably as opposed to the "developmental research" that had dominated the AEC program.
>
> Several attempts were made by RES to define "confirmatory research." The need for definition, however, was in effect obviated by the issuance of a directive that the research must be responsive to and, except in special circumstances, initiated by "user" needs. Users were considered to be principally the other NRC program offices but to include also the Commission, Atomic Safety and Licensing Boards, Atomic Safety and Licensing Appeal Board, ACRS, Congress, the technical community, and the public.
>
> In its reviews of the NRC Safety Research Program during the past several years, the ACRS has found it increasingly difficult to distinguish between "confirmatory research" and "research to improve reactor safety." As a result of the accident at Three Mile Island Unit 2 (TMI-2), the distinction has become even less obvious and, clearly, less important. Several of the proposals by NRC and others resulting from the accident at TMI-2, which have now become matters for licensing decisions, were previously relegated to the category of "research to improve reactor safety" and were inadequately funded as a separate program area.
>
> The ACRS believes that a rather unfortunate distinction has been made between "confirmatory research" and "research to improve reactor safety". Although the latter has consistently been assigned a high priority by RES in its requests for funds, it has been difficult to provide funds for "research to improve reactor safety" under circumstances where this might have resulted in some reduction of funds for an item of "confirmatory research" believed by one of the "user offices" to be directly related to their own real or perceived needs. It must be noted, however, that this attitude may be changing, and

may be expected to change further in the aftermath of the TMI-2 accident. Research on alternate decay heat removal concepts (one of the projects identified in NUREG-0438, "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants," as "research to improve reactor safety") has already been started, using "confirmatory research" funds, in response to a request from a "user office". In addition, the concept of a vented and filtered containment has received prominent mention in NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report", so that research in this area might also be the subject of a "user request" and, qualify for support from "confirmatory research" funds.

The foregoing comments indicate that the distinction between "confirmatory research" and "research to improve reactor safety", is neither very clear nor useful, if indeed there ever was a legitimate distinction. It is similarly artificial to stipulate that in its research to improve reactor safety the NRC should not make use of experimental studies where such studies would seem to be the most advantageous means to follow. The ACRS suggests that the Congress review the legislative charter of the NRC research program and eliminate this distinction. Inasmuch as the purpose of the NRC is to ensure safe design, construction, and operation of nuclear power plants, it should follow that any research is legitimate that contributes to that purpose without compromising the credibility or independence of the NRC. The ACRS urges consideration of this matter by the Congress.

Question 2

Would you cite examples of the kind of research ACRS was referring to in saying that the NRC should reduce sharply confirmatory research in areas where current regulatory requirements provide adequate protection to the public?

Response

Some of the areas in which the ACRS believes that the NRC Staff has adequate information and that additional research may not be necessary are as follows:

- Large-break Loss-of-Coolant Accident (LOCA) research intended to demonstrate, within reasonable limits, the effectiveness of PWR emergency core cooling systems. (Subsequent to the TMI-2 accident, more emphasis and increased attention has been given to small-break LOCA problems.)
- Loss-of-Fluid Test (LOFT) Program - This was originally intended to look at large-break LOCA problems. Since necessary information has been obtained in the large-break LOCA area, coupled with the fact that the emphasis has been shifted toward small-break LOCA problems, the LOFT program is no longer cost effective.

- Some of the work in the Power Burst Facility and other facilities such as NRU (Atomic Energy of Canada, Ltd., Test Reactor), which consists of in-pile tests of fuel rods, development and assessment of fuel codes, and basic studies of cladding behavior in fuel bundles during normal operational transients and design-basis accidents, represent an area where information now available seems adequate for any high priority regulatory need.

- Some work associated with improvement in the modeling of meteorological dispersion/transport of fission products. The ACRS believes that such work should be pursued only to the extent that the quality of the related input data justifies.

Question 3

Your report also stated that the NRC research response to the TMI-2 accident has largely been the implementation of remedies to specific issues subject to early identification and resolution, but that there has been a lagging, incoherent approach to more general and probably more important issues. Would you elaborate on these more general issues and suggest how they should impact NRC's research?

Response

The ACRS' report noted that the overall NRC response, rather than the research response, has largely been on remedies to specific issues subject to early identification and resolution. In its report to the Congress on the FY 1982 NRC Safety Research Program (NUREG-0751), the ACRS recommended that the following areas be given higher priority:

- The role of control systems in safety.

- Plant operational safety, including system behavior as a function of design.

- Reliability analysis of existing plants, including emphasis on the more detailed understanding which might prevent many operating occurrences which have potentially serious implications.

- Improved, more reliable shutdown heat removal systems, including dedicated and bunkered systems.

- Studies of degraded core and core melt accidents with emphasis on the conceptual design and evaluation of features to mitigate such accidents.

- Studies of the physical and chemical behavior of fission products in post-accident environments.

- The early development of an approach to supplement or replace the single failure criterion.
- Other matters relevant to setting the principal design basis for plants to be constructed.

Further, in its report to the Congress on the FY 1983 NRC Safety Research Program (NUREG-0864), the ACRS identified several issues, as given below, that should receive higher priority in the NRC Safety Research Program:

Human Factors

The role of the human in reactor safety has received considerable emphasis since the TMI-2 accident. The NRC Safety Research Program in this area has been focused in large part on improvements in control rooms, in the display and monitoring of safety parameters, and in operator training and procedures. The ACRS believes that the NRC should take the initiative in defining a broadened research program on human factors, one which allows industry the lead role where industry has accepted and is meeting the responsibility of establishing an appropriate program. Among the matters which this broadened program should include are the following:

- Improved bases for decision-making in the control room during complex transients, including more comprehensive plant procedures and development of a diagnostic adjunct to the Safety Parameter Display System (SPDS).
- Development of an improved approach for judging the effectiveness of licensee management.
- Improved understanding of the impact of maintenance and testing activities on plant safety.

Control Systems and Plant Operational Safety

In its previous reports to the Congress and the Commission (NUREG-0751 and NUREG-0795), the ACRS has recommended that higher priority and greater funding be given to the role of control systems in safety. Related matters of plant operational safety, such as instrumentation systems for the detection of inadequate core cooling, also deserve additional work. The NRC has initiated some effort on the more general issues involved, but the ACRS believes that higher priority and funding levels are warranted than those proposed by the NRC.

Sabotage

In recent years, the NRC has imposed fairly extensive requirements on access control for nuclear power plants for purposes of security. However, a thorough evaluation of possible design approaches to help protect against sabotage by an insider has not been undertaken and plants in the U.S. do not now include some protective measures which are required in several European countries. The ACRS believes that study of further protective measures against sabotage warrants high priority and the research necessary to assist NRC decision-making should be expedited.

Aging

As nuclear plant operating life advances, some degradation in its equipment and systems must be expected. Pressure vessel embrittlement has been recognized for many years to increase with accumulated fast neutron fluence requiring adjustment of the operating temperature and pressure limits to assure rupture resistance. The frequency of steam generator tube leakage also increases with operating life. Organic coverings of electrical and control cabling, elastomer seals on equipment closures, and potting compounds for electrical connections are all known to deteriorate under extended exposure to humidity and temperature conditions and low level gamma and beta irradiation. Cyclic operating conditions affect the performance reliability of valves, motor operators, ventilation machinery, and comparable equipment. Emergency diesel power equipment and DC power sources also are subject to degradation with operating use. The safety of nuclear power plants depends on making certain that such degradation with age is recognized and accommodated before it can cause significant reduction in safety.

There is inadequate knowledge regarding the effectiveness of the surveillance programs related to aging and the effects of operating transients on equipment that has deteriorated with age. As highlighted by the recent concern for thermal shock on some older pressure vessels, it appears important to apply some research effort to improve understanding of the safety significance of loss of capability through such aging. The NRC should initiate a comprehensive, systematic investigation of safety-related effects of aging for Light-Water Reactors (LWRs).

Quality in Design and Construction of Nuclear Facilities

Construction and operating experience continue to highlight the need for added emphasis on improved quality in LWRs. Studies should be initiated to examine the potential for reducing the incidence of design and construction errors. These studies should take account of the experience and practices in other high technologies and in other countries. In addition, studies are needed to determine the ability of operating plants to operate safely in spite of various kinds of design and construction flaws.

Standardization

There is a growing opinion in favor of standardization of LWRs as it affects both the safety and the reliability of nuclear power plants. However, the term standardization is ill-defined and, if standardization is pursued, it should be done in a way that ensures improvements. The ACRS recommends that the NRC initiate the studies necessary to formulate a regulatory position on this matter, including the possibility of recommending changes in the antitrust laws, if necesssary.

Some Other Matters Warranting Increased Emphasis

Other matters warranting increased emphasis, priority, and funding in the NRC Safety Research Program include the following:

- Improved approaches to more reliable shutdown heat-removal systems for both the reactor and the containment.
- Expanded knowledge about the likelihood and effects of severe water hammer during a wide range of possible transients and accidents.

Question 4(a)

ACRS singled out the "Accident Evaluation and Mitigation" decision unit as warranting lower funding than proposed due to the absence of a program that is focused to meet NRC needs. Is this recommendation based on a lack of confidence in particular program elements?

Response

The ACRS recommendation in NUREG-0864 to reduce the funding in the "Accident Evaluation and Mitigation" decision unit by $6 Million is based on the following considerations:

- A clear need has not been established for Phase II of the Severe Fuel Damage Test Program in the Power Burst Facility (PBF) and the overheating tests in the NRU (Atomic Energy of Canada, Ltd., Test Reactor).
- The separate effects experiments in the Annular Core Research Reactor (ACRR) are too detailed and too long term to provide vital information for the NRC rulemaking activity associated with severe accidents. They should be deferred (see the response to question 5).

- The experiments underway and contemplated on the fuel melt behavior are elaborate and expensive and there has been no evidence indicating that they adequately reflect the processes that will be critical in defining the rate of core damage evolution. Therefore, these experiments should be deferred until a more coherent analysis has been performed to define the most probable accident evolution, and to identify critical information required to assess accident progression.

- Not enough attention has been given to the question of how much detail is desirable or feasible in the code development work associated with fuel melt behavior. Without careful planning it is possible for the experiments to become a vehicle for code development while the codes become a tool for describing the experiments.

Question 4(b)

What would you include in the "substantial portion of the longer-term experimental and code development work" which ACRS said could be eliminated in a better-focused program?

Response

Regarding the elimination of code development work, see 4(a) above. In addition, the ACRS believes that it is virtually impossible to develop the detailed information and computer codes needed to model the processes leading to all of the important core melt scenarios; yet the research on the behavior of damaged fuel proposes to do essentially this.

Question 5

Why does ACRS feel that the separate effects experiments on the behavior of damaged fuel in the Annular Core Research Reactor are too detailed?

Response

The separate effects experiments in the Annular Core Research Reactor are intended to provide information regarding the effects of various parameters such as pressure on debris coolability. These experiments are examples of experiments that are too detailed and too long term to provide vital information for the proposed NRC rule on Severe Accidents. Decisions as to how to proceed with the rule will be based in part upon the results of the integral tests, and the separate effects test results will only provide confirmation. The ACRS, therefore, believes that the separate effects experiments can be deferred in favor of higher priority work.

Question 6

ACRS expressed some concern about the goals of the German Upper Plenum Test Facility and their priority in NRC's safety research. In what ways do you recommend that NRC attempt to redirect its portion of the expenditures in this program?

Response

The 2D/3D program is an international effort involving Japan, the Federal Republic of Germany (FRG), and the United States, and was begun when LWR safety research was preoccupied with large-break LOCAs. This program consists of two components: (1) The FRG Upper Plenum Test Facility (UPTF), (2) the Japanese Cylindrical Core Test Facility (CCTF) and Slab Core Test Facility (SCTF). The U.S. contribution to the 2D/3D program is the supply of experimental measuring devices, and analytical support by the NRC.

The FRG UPTF will presumably provide some information relating to special questions regarding large-break LOCAs. One is the so-called Emergency Core Cooling (ECC) bypass question which ceased to be of concern many years ago. Further, a large part of the UPTF program in FRG relates to hot-leg injection, which is a special feature of the FRG PWR design, and may not provide helpful information to the U.S. PWR reactors which are designed to inject ECC water into the cold leg. The ACRS believes that the FRG UPTF promises to be a very costly facility with insufficient return in pertinent data. In light of the above, the ACRS recommended in NUREG-0864 that the NRC attempt to redirect its portion of expenditures to other research which will provide a more significant contribution to thermal-hydraulic problems important to the U.S. reactor safety.

The ACRS has not made any specific recommendations with regard to redirection of these particular funds.

Question 7

What concerns led to the ACRS recommendation that the program on fission product release and transport be subjected to careful peer review?

Response

Fission product release and transport is a major element in accident analysis and site evaluation. Radioiodine releases, for example, are major potential contributors to accident dose levels because of iodine retention in the human thyroid.

The release and transport of radionuclides during and following a reactor accident involves a complex interplay of the accident scenario, performance of engineered safety features, and environmental factors which impact on the physical characteristics and transport phenomena for the fission products. An admittedly oversimplified and conservative source term has been used as the basis for safety evaluations since the publication of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," by the U.S. Atomic Energy Commission in 1962. These conservative values are embedded in a number of NRC regulatory requirements and guides.

In the past, iodine was assumed to be released from a damaged core as elemental iodine due to its volatility (see TID-14844) and a fraction as methyl iodide, an organic form that cannot be readily removed from an airstream. These assumptions were recognized as being conservative but prudent as far as protecting the public. On several occasions in the past there have been core damage accidents and the observed iodine releases were significantly less than assumed in the analyses.

Recently doubts have been raised regarding the form of iodine that would be released during an LWR accident. Evidence suggests that iodine takes a form that is less volatile and more soluble in water than elemental iodine. If the iodine does take this form, the end result is that the risk to the public from radioiodine would be significantly lower following an accidental release than many people now predict, and that some other radionuclide might become a dominant concern.

After discussing this phenomenon with interested and knowledgeable parties, including FRG authorities who have done and are doing considerable work in this area, the Nuclear Regulatory Commission directed its staff to conduct a reevaluation of fission product release and transport as used in the theoretical analyses for accident assessment and reactor siting.

Because of the importance and complexities of this research, the ACRS wants to ensure that the NRC staff has the best plan to assure unambiguous results and that the research results are of the highest quality and take into account the extensive work in this area by other knowledgeable individuals and organizations. A peer review with breadth, as well as depth, is needed to consider accident scenarios as well as fission product behavior. For example, are there any scenarios of significance for which various source reduction mechanisms would not be operative?

Question 8

ACRS suggested that even with a complete set of input data to describe a given plant, an evaluation using the methods of the Seismic Safety Margins Research Program (SSMRP) may be so beset with uncertainties as to provide little basis for licensing decisions beyond the judgments already being made. Would you explain the basis for this concern?

Response

In applying the methodology being developed in the SSMRP to a plant, the analyst must first obtain a detailed knowledge of the plant structures and components. This can be in itself time consuming and expensive. The SSMRP methodology requires input parameters, (other than those relating to the plant configurations) which, within the present state-of-the-art, can be determined only with large uncertainties. Examples are the return period for large earthquakes, the response spectrum for these earthquakes, the fragilities of plant components, and the dynamic behavior of materials. The SSMRP methodology is in itself unverified and is likely to prove difficult to verify. The ACRS was concerned that these uncertainties may be so large as to prevent the results of the analysis from being useful. The ACRS believes that the program plan and objectives should be reviewed critically, by a group including experts in seismic engineering and seismology, and the program redirected as necessary.

Question 9

The ACRS recommended that human factors research ought to be redirected from control room design to those areas already identified as significant contributors to risk. What specific areas should receive this redirected emphasis?

Response

Most human errors occur outside the central control room. They usually are not caused by an operator misinterpreting a piece of information received from a poorly designed gauge or meter. Errors as a result of improperly designed controls result in only a very small fraction of the human errors reported. Research which could have a greater potential to reduce the human contribution to risk includes cognitive recognition of abnormal conditions and development of plant procedures for their control. A significant industry effort is underway to develop technical guidelines for plant emergency procedures. The ACRS believes that this effort will result in an improved ability of licensees to manage abnormal and emergency plant conditions. Research into the Safety Parameter Display System should be directed toward optimizing the design and expanding it to include diagnostic capability. Research into developing objective and measurable criteria for judging utility management effectiveness should also be performed. Finally, research efforts should be increased to develop requirements and guidelines for improving equipment design, test design, and maintenance procedures, incorporating human factors principles.

Question 10

The ACRS cited a need for more formal interaction with EPA on uranium mining and milling? What led to this concern.

Response

The EPA has legislative mandates for developing radiation protection standards for uranium mining and milling. These standards have an impact upon the NRC's licensing role in this area. For this reason, the ACRS recommended that the NRC Staff establish formal channels for communication and interaction with the EPA. The NRC Staff is actively working to implement this recommendation.

Question 11

In reference to Earth Sciences research, ACRS recommended examination of the state of knowledge of seismology and geology, keeping in mind problems which arise in the application of past regulatory practice. Would you explain what problems are referred to here?

Response

While large uncertainties persist in the estimation of the likelihood of increasingly severe earthquakes, there have been considerable increases in the understanding of some of the underlying causes of earthquakes as well as a change in viewpoint concerning the source and the likelihood of earthquakes recurring in the future at locations where they have occurred in the past, east of the Rocky Mountains. Furthermore, improved knowledge of attenuation and stress states is developing. While some of this knowledge is being factored into current seismic reviews, these reviews are constrained, in part, by difficulties arising in assigning boundaries to tectonic provinces and by other facets of adhering to Appendix A of 10 CFR Part 50, the NRC General Design Criteria. The ACRS believes that a broad reevaluation of the overall state of knowledge and regulatory practice could be of value.

Question 12

The ACRS recommended that NRC review the risk-limitation effectiveness of general design criteria, regulatory guides, and standard review plan. Would you clarify the effort that is being suggested here?

Response

At present the NRC is attempting to establish priorities for its efforts to resolve safety issues, and to assess the efficacy of proposed resolutions to the issues. All of the current regulatory requirements are the result of attempts to address specific safety issues. However, most of these requirements, including many parts of the Task Action Plan resulting from the TMI-2 accident, have not been subject to the review that is planned for new requirements. Although it may not be reasonable to review all of the current requirements as thoroughly as is planned for the resolution of current safety issues, some effort should be made to reconcile the differences in the treatment of new and old requirements.

Question 13

You note in your testimony that the Commissioners should help establish research priorities by addressing questions relating to the respective roles of NRC, DOE, and industry in safety research. Would you describe ACRS' view of the appropriate roles of these segments?

Response

The ACRS believes that there are complementary roles for both NRC and DOE in research to improve light-water reactor safety, and that aggressive programs should be pursued by each agency with appropriate coordination. It is both desirable and appropriate for the NRC to conduct research on safety concepts, but their development and implementation should be carried out by the nuclear industry or DOE. In its 1978 report to the Congress on the NRC Safety Research Program (NUREG-0496) the ACRS noted that:

> "The ACRS believes that the development, testing, and proof of efficacy of new or improved safety systems should not be the responsibility of the NRC, but should be conducted by the nuclear industry or DOE.
>
> "However, the ACRS believes that it is a proper and even necessary function of the NRC to perform or sponsor research on concepts that, if developed and implemented by the appropriate bodies, could lead to improvements in safety".

The ACRS believes that the NRC cannot and should not perform research in each and every subject area. By asking the right type of questions, NRC should be able to take advantage of the research being done by industry and DOE in several areas. Further, the ACRS believes that the NRC has the authority and mechanisms to get industry to perform research in certain areas. Industry has been performing research, as necessary, to answer some questions raised by the NRC in the licensing process.

The ACRS believes that prior to initiating research on its own, the NRC should ask itself the following questions:

- Will the research serve the needs of the NRC in its mission to protect the public health and safety?
- How will the research sponsored by the NRC relate to that being done by others? Will it duplicate research being done by other comparable agencies and industry? Will it be needed to supplement or confirm work being done by others?

- When should research be done by NRC, when by DOE, when by industry, and when and how by a combination of these?

- Should some research, particularly that which involves large scale component testing or the application of existing methodology, be the responsibility of the industry rather than of the NRC?

The Electric Power Research Institute (EPRI) has a base research program with a funding level of about $59 million for FY 1982; and in FY 1983 it plans to spend about $65 million. In addition, it has a progam in FY 1982 with a funding level of about $23 million to perform research in the areas of steam generators, relief valve testing, BWR pipe cracking, etc. The ACRS believes that the NRC should take advantage of research being done by EPRI in these areas.

In its previous reports to the Congress and the Commission (NUREG-0751, NUREG-0864, NUREG-0699, and NUREG-0795) the ACRS made several comments regarding research that should be done by the industry or DOE. Some of these are as follows:

- The fire protection and qualification testing programs are in support of developing regulatory guides and standards. In addition, tasks within these programs have been carried out at the request of the Office of Nuclear Reactor Regulation of the NRC to investigate existing plant installation for adequacy. The ACRS believes, however, that the expense of fire replication tests is far too great for the information to be obtained. As a result the ACRS does not support this particular part of the program. Industry should be encouraged to perform more confirmatory testing of fire protection concepts.

- The nature of many of the problems related to structural safety is such that special attention should be given to the question of whether the needed research should be done by the NRC or by the industry.

- The Office of Nuclear Regulatory Research (RES) of the NRC should maintain cognizance of the structural research being done by industry and should be in a position to utilize the results of it to the greatest extent practicable.

- One of the larger components of the Operating Effects on Materials Program is the examination of the Surry steam generator by the Battelle Northwest Laboratory. This provides a unique facility for needed NRC work on the verification of nondestructive examination (NDE) techniques. However, it could also be very useful to industry for activities such as training of NDE operators, chemical cleaning studies, etc. The ACRS suggests that the NRC limit its exclusive control of the facility to a short period and look into then transferring the facility to private management so that a wider range of work of value to the industry can be performed with this steam generator.

- The RES Staff should evaluate whether it is appropriate for NRC or industry to finance some proposed work in the Mechanical Engineering area. Snubber qualification, for example, might more appropriately be sponsored by the industrial suppliers and the industry users.

- The NRC Staff should not attempt to develop methods of analysis appropriate to each of the many containment types. The responsibility for computing and justifying containment behavior and capacity should be placed on applicants and licensees, while NRC should develop the expertise needed to direct and review the licensees' or applicants' programs.

- NRC should vigorously pursue industry contributions for the proposed Semiscale, MOD-5 program.

- The code development work in the Fuel Melt Behavior area is expensive. Attention should be given to the question of how much detail is desirable or feasible. The NRC should coordinate, insofar as feasible, its code development work in this area with that of DOE and of industry.

- The ACRS recommends continuing careful integration of the NRC work related to hydrogen in the Accident Mitigation area with the extensive investigations being carried out by the industry.

- It is not necessary that all of the safety-related research on Liquid Metal Fast Breeder Reactors be conducted by the NRC. Rather, it is appropriate that, as a logical and essential part of its development programs, a major portion of this work be done by DOE. Even so, it will still be necessary for a significant amount of research to be conducted by the NRC to develop criteria, to determine which safety-related questions must be answered by DOE research, and to be able to evaluate and confirm the answers or results from DOE research. The NRC program should be coordinated not only with research being performed by DOE but also with that being performed by foreign groups.

Department of Energy
Washington, D.C. 20585

June 28, 1982

Mr. John V. Dugan, Jr.
Staff Director
Subcommittee on Energy Research
  and Production
U.S. House of Representatives
Washington, D. C.   20515

Dear Mr. Dugan:

Enclosed are answers to your questions on the Nuclear Regulatory Commission research program addressed to Gordon Chipman, Deputy Assistant Secretary for Nuclear Energy, following the May 19 hearing of the Science and Technology Subcommittee on Energy Research and Production.

If you have any questions, please don't hesitate to call Diana Diamond or Tom Pretorius of my staff at 252-4277. They will be happy to assist you.

Sincerely,

Thomas H. Pretorius

for Robert G. Rabben
Assistant General Counsel
for Legislation

Enclosure

Q1. Do you feel that there is unnecessary duplication of effort between NRC and DOE research efforts on breeder reactors? If so, please cite specific examples.

A. There is no significant or unnecessary duplication of research efforts between the DOE or NRC concerning the breeder reactor. The major effort in breeder reactor research is with the Department of Energy. The Nuclear Regulatory Commission has a relatively small breeder reactor research effort since it devotes most of its research efforts to light water reactor technology.

Q2. There has been previous concern about the amount of interaction of NRC with DOE and EPA. Have DOE and NRC contacts, especially in the area of waste management and reactor safety, been increasing?

A. In terms of waste management, our contacts with the NRC have been increasing mainly in such areas as site characterization work and in design documentation for the repository. The reactor safety activities of NRC and DOE have always been well coordinated, but because of the limited scope of the DOE LWR safety program, the contacts have been limited to a small segment of the NRC organization. With the enactment of the Nuclear Safety Research Development and Demonstration Act of 1980, the Department has convened working groups to identify program needs in all areas of reactor safety. The NRC has been active on these working groups and as a result of this, the amount of interaction with DOE has increased significantly.

Q3. What are the differences in the DOE and NRC modeling efforts for the high level waste program?

A. The major difference in the DOE and NRC modeling efforts for the HLW program concerns the depth of modeling detail. The DOE program includes three levels of models:

- An overall systems model
- Three subsystem models (package, repository and site)
- More than 30 individual process models

The NRC's modeling efforts are focused on the more general subsystem models, which describes the physical phenomena to a first-order approximation. The DOE is also developing these subsystem models and is developing complex models of individual processes (such as corrosion, rock-mechanics effects on groundwater flow, etc.) important to the isolation of wastes. A small number of the subsystem simulations are checked in detail by iterative application of the individual process models. This approach attempts to strike a balance between efficiency and accuracy in modeling applications.

Q4. A clear purpose of the NRC research program is to support the licensing activities of the Commission. One of the issues with the High-Level Radioactive Waste bill is how long the licensing process would take for a repository. Is there additional research that NRC could do that would enable them to shorten the licensing process?

A. NRC should concentrate their research on ways to determine how compliance with the major performance objectives that NRC may establish, e.g.:

- 1,000 year containment time
- one part in 10^5 annual release rate from engineered system
- (EPA specified) cumulative quantity of radionuclides entering accessible environment

NRC's research should emphasize testing of reference waste forms selected by DOE, i.e., spent fuel and borosilicate glass, rather than investigating properties of alternative waste forms.

Q5. Do you believe that NRC should do any additional research to establish the criteria for high level waste forms and packaging?

A. The main need in this area is to establish methods for showing compliance with the requirements in the proposed regulation, 10 CFR 60. Methods are needed to predict the performance of waste forms and waste package for very long times based on relatively short-term tests (one to 10 years). We suggest NRC instead consider other supporting information such as the naturally occurring reactor (OKLO) and underground nuclear device tests and focus their attention on these area rather than on development of "better" waste forms and packages.

Q6. Is there unnecessary duplication of effort between NRC and DOE research efforts on high level waste management? If so, please cite specific examples.

A. We would like to highlight two specific research areas in which there may be some duplication. As discussed in response to Question 3, the NRC is developing a suite of general subsystem models to simulate the post-closure performance of a repository. The DOE is also developing a suite of these more simple subsystem models, so there may be some duplication at this intermediate level of model development. But this may be appropriate in view of NRC's responsibility to confirm the DOE's predictions. We believe there is virtually no duplication in the field of detailed, individual-process model development.

We also feel that some of NRC's research efforts on borehole sealing may be premature. We understand that NRC is already sponsoring field tests of sealing materials in granitic rocks, although DOE's siting program to date has concentrated on salt, basalt, and tuff.

ELECTRIC POWER RESEARCH INSTITUTE

EPRI

June 23, 1982

Honorable Marilyn L. Bouquard, Chairman
Subcommittee on Energy Research and Production
Committee on Science and Technology
U.S. House of Representatives
Suite 2321, Rayburn House Office Bldg.
Washington, D.C. 20515

Dear Chairman Bouquard:

On May 19 I testified before your subcommittee on those topics which you requested in your invitation letter. John V. Dugan, your subcommittee staff director has asked for clarification of points arising from that testimony in his letter of May 25, 1982. This letter addresses those points of clarification requested in Mr. Dugan's letter.

The amount of research being done by the federal government is judged by us as essentially correct, though we encourage focusing the work better, we encourage coordinating the NRC research with that being done by industry and urge NRC apply some of the excellent research results of the past in the licensing process. In answering the numbered items in Mr. Dugan's letter we begin with a paragraph restating the question asked and answer it in the following paragraph.

Response to Further Questions from Energy Research and Production Subcommittee NRC Research Program on the Testimony of May 19, 1982

1. We understand that coordination by the NRC of its research programs with related industry efforts has been increasing. Would you enumerate your cooperative programs with the NRC?

   Coorperative programs include FLECHT-SEASET (Westinghouse-NRC-EPRI), BWR FIST, THERMAL SHOCK (ECC MIXING), BWR REFILL-REFLOOD and STEAM GENERATOR TRANSIENT TESTING (NEW).

2. NRC testified that additional research is necessary before results of the LOFT test program can be used to modify significantly Appendix K of 10CFR50. Do you agree with this assessment?

   No. Improvements would be needed in currently used computer codes to apply the knowledge gained from the test data we have from LOFT. Other supporting data is also available from programs such as FLECHT-SEASET.

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3. At the hearing Dr. Siess was asked a question about seismic designs. Dr. Siess commented that several non-nuclear powerplants have withstood earthquakes without damage, but he felt nuclear powerplants may not do as well because of the additional stiffness of snubbers, etc. required in the name of seismic safety. In other words, the regulation may be making nuclear powerplants less safe, not safer, in the case of an earthquake. Do you feel this question is being adequately addressed in the NRC research program?

   Additional stiffness introduced by snubbers will be beneficial by mitigating piping response in a seismic event. However, during normal operation where piping systems undergo many cycles of heat up and cool down, additional pipe support stiffness (introduced by snubber lock-up under low frequency thermal cycling in the event of snubber malfunction) will induce undesirable thermal loads and thereby introduce more severe fatigue considerations. The balance between a stiff versus a flexible piping system to ensure safety under seismic as well as other loading conditions is adequately addressed in the NRC long-range research program. Results and test data obtained through EPRI's program will also complement NRC's objective for an optimal piping system with maximum overall safety.

4. The NRC Long Range Research Plan does have a strong emphasis on continuing current programs. Too frequently, large facilities are operated past their time of usefulness because of the tremendous investment of money and people that they represent. In your view, how closely wed is the NRC's research plan to large-scale facilities?

   Very closely - both to facilities and to analysis groups, e.g., Semiscale, PBF, LOFT, and the RELAP and TRAC computer code groups at INEL and LASL.

5. You testified on the importance of source term definition for evacuation regulations for severe accidents. Please be more specific on what research results would allow for a change in the regulations on evacuations and warnings.

   Regulatory guidance for protection of the public in the event of a severe reactor accident has substantial emphasis on emergency response actions, including early warning, evacuation, and the use of thyroid blocking agents to prevent radioiodine uptake. However, such guidance is based on very conservative analytical projections of off-site radiation exposure conditions which, in the light of careful reevaluation of current experience and experimental and theoretical evidence, are overstated in terms of radiation levels, quantities of radionuclides present, effected distance from reactor containment and duration of time between accident initiation and releases of radionuclides to the environment.

Research which would allow for more realistic determination include:

- Improved data on the physical and chemical behavior of volatiles escaping from damaged fuel elements under actual, expected plant system conditions.

- Better modeling of the transport of such volatiles from the core to be released off-site, using improved thermodynamic codes.

- Reevaluation and updating of present assumptions on failure of containment systems, including review of realistic failure modes and performance.

- Reevaluation and updating of currently assumed reactor accident scenarios, including "dry" accidents.

- Reevaluation of present assumptions on environmental transport from the point of release to off-site areas, identifying the degrees of conservatism involved.

- Initial evidence indicates that the above research will yield the following results:

    1. Fractions of radioiodine and radioactive particulates released to the environment are substantially reduced, perhaps by a factor of 10 to 100.

    2. The time before release of noble gases and reduced fractions of volatiles is extended from minutes to hours or even days.

The combination of these results would substantially alter the technical basis from which present evacuation and warning requirements were derived, and should therefore lead to reduced requirements for warning systems, a small emergency planning zone, and less emphasis on evacuation. The need for pre-distribution of potassium iodide tablets to the public, which is questionable even under the present basis, would disappear. These changes, if proven, should not be set aside in favor of maintaining the levels of emergency preparedness now stipulated in NRC requirements. The cost in terms of public apprehension and misunderstanding, as well as the personnel and equipment cost to the industry for unneeded capability, are too great. Hopefully, this research will be completed on a sufficiently urgent schedule to effect realistic emergency plans promptly.

6. Can you give an example of where the research results have improved or enhanced the safety of nuclear power plants?

To answer properly the very broad question of the applicability of research results to improve reactor safety can benefit by stating some historical context. Much of the extensive research work that was carried on for years at the National Laboratories was directed almost solely to reactor safety. For many years, reactor safety research was a line item in the old Atomic Energy Commission budget as approved by Congress. Perhaps the most striking example is the early BORAX experiments. In these experiments, a small prototype of a boiling water reactor core was pulsed by having its control rods drawn out rapidly as power was allowed to increase very rapidly in an uncontrolled fashion. In test after test the reactor showed how the heating and expulsion of water from the core as the power went up caused the power to turn around and decrease sharply, thus limiting the energy release and shutting down the reactor. These safety studies showed how difficult it was to cause an accident in a boiling water reactor, and served as a basis for the safety design of future boiling water reactors. Later, a long series of tests in a specially built reactor called SPERT was conducted to provide much more detailed and quantified safety information for pressurized water reactors.

A current issue is pressurized thermal shock. This involves the fact that the steel reactor pressure vessels become more brittle over their lifetime as they are exposed to intense bombardment by neutrons from the reactor core. Over the past three decades, extensive experiments have been carried out in which samples of reactor pressure vessel steel have been irradiated to very high neutron exposures in order to simulate the condition of the reactor pressure vessel metal at the end of a reactor vessel lifetime. The data obtained from this research show that reactor vessels can withstand a lifetime of exposure and retain their integrity. Recently, NRC and the industry have been examining the available data, and considering both routine operation and unusual accident scenarios to determine whether or not a transient could inflict greater stresses on the embrittled metal than had been predicted when the reactors were designed. The available research base is sufficient to give confidence that reactor vessel safety can currently be assured, but further research is needed to reduce the margins of uncertainty on experimental data so that more precise predictions can be made.

Another example is the wide variety of experiments on the interaction of the reactor cooling water with overheated fuel and fuel-cladding material. These studies range all the way from normal operation to the most critical hypothetical accident scenarios. Early experiments included the spraying or dropping of liquid metal into water and steam mixtures to determine metal water reaction rates. These experimental data are then used in detailed computer calculations to determine how reactor systems may actually behave.

A more recent example is the missile test series directed by EPRI. In this research, a wide variety of missiles were catapulted against barriers that would simulate the reactor containment building. The missiles ranged from telephone poles (as if uprooted in a tornado), and steel bars and pieces of turbine casings which, it was predicted, could be broken loose in the event of a severe disintegration of the steam turbine of a nuclear power plant. These tests provided data which showed the adequacy of the safety design of the containment building, and also showed how future containment buildings could be designed more efficiently. These same types of tests were used to study the safety of spent-fuel casks, the shielded containers in which highly radioactive fuel elements are transported from the reactor to a reprocessing plant. These tests firmly established the adequate safety of the design, and provide a basis for regulations which assure the safety of the public during the transportation of highly radioactive materials.

Recent research has developed the field of probabilistic risk assessment (PRA). This powerful tool enables designers and regulators to make determinations about the safety performance of reactor systems, and in this way to determine not only safe ways to design them but to develop a quantitative measure of how safe the systems are. Using PRA one can make determinations as to how much safety should be designed into a particular system as well as determine when additional engineering effort to increase safety has reached a point of diminishing return, and where those resources can be spent improving the safety of other parts of the system.

7. Can you give a successful example of where research has resulted in a softening of overly conservative requirements?

There have been no cases where overly conservative regulations have been softened. Much more often the result has been the recognition on the part of regulatory bodies that new restrictive actions need not be taken. In most cases, potential regulations are not published as drafts in the Federal Register because discussions at the NRC staff level and reviews of the research literature cause the NRC staff to make changes in proposed drafts or drop the proposal for new regulations before an inordinate amount of time and effort has been spent on it. The tornado missile experiments conducted by EPRI are an example of a case in which some NRC thinking about increased design requirements had been undertaken and was dropped when it became obvious that the experiments showed that the containment structures were safe. Another example, is a series of tests on model containment structures by EPRI to determine the response to seismic events which has provided NRC with a basis for not needing an increase in regulatory requirements.

NRC, EPRI, and vendor research have demonstrated that fracture of most nuclear plant pipes in an abrupt guillotine manner is a very improbable event. The impact of this will most likely be to show additional restrictions to handle hydrodynamic loads following guillotine large break LOCA are unnecessary. This item is still pending resolution and could be influenced by NRC, EPRI, and German research showing that the results of these hydroloads is less severe than originally thought.

The importance of research in two crucial areas at the present time falls into this category. One is the entire area of the fission product source term that could result from serious core damage accidents. There is a wealth of experience in the nuclear industry, as well as in chemical process industries and others, that shows that the design calculations used in reactor accident calculations, emergency planning and overall safety assessment are very conservative. It is EPRI's goal to provide definitive research and documented evidence for NRC with the necessary basis for impact on past and potential regulations which has resurfaced upon learning "new" information.

8. There has been a fair amount of discussion about hydrogen embrittlement, steam generator leaks, and thermal shock or overpressuring action. Do the research results indicate that the utilities should be concerned about these issues shortening the expected lifetime of a nuclear power plant? Do you feel that the NRC research program will resolve these issues? When?

The issues are of concern because of implied NRC regulatory action to impact plant licenses. Issues, such as those you have mentioned, are bound to arise in the course of time as we learn more but should be placed within the context of overall risk.

Hydrogen embrittlement is a well known metallurgical degradation of certain steels which under some circumstances can contribute to structural cracking. It has been observed as a possible contributor in a French pressure vessel, but attributed to fabrication techniques which were unique to the French operations. Hydrogen embrittlement is not believed to be of importance in U.S. Light Water Reactor pressure vessels for the lifetime of the plants.

Steam generator leaks are a serious operational problem in the nuclear industry. At EPRI we have a Steam Generator Owners Research Program which has addressed some of the earlier perceived difficulties such as denting. New aspects of steam generator reliability such as stress corrosion cracking convinces us that the steam generator performance problem will require considerably more work before the problem is under control to our satisfaction. The new NRC program at Battelle Northwest which is being developed in cooperation with the ongoing EPRI program is expected to help solve this problem.

EPRI is pursuing research on the subject of pressurized thermal shock to improve our knowledge base as discussed under Question 6.

To summarize, these issues should be placed within the context of overall risk. We believe it is NRC's responsibility to insure that only risk significant issues are addressed and that NRC and industry resources are not wasted on other than risk significant issues.

Thank you very much for the opportunity to share our views with you. We hope that they will prove useful. If we can be of any further assistance, please do not hesitate to ask.

Best regards,

John J. Taylor, Director
for Nuclear Power

JJT:bg

cc: John V. Dugan, Staff Director
Subcommittee on Energy Research and Production

Floyd Culler, President
Electric Power Research Institute

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 27 1982

The Honorable Marilyn L. Bouquard
Chairman, Subcommittee on Energy
  Research and Production
Committee on Science and Technology
United States House of Representatives
Washington, DC 20515

Dear Madam Chairman:

In response to your letter dated July 2, 1982 posing additional questions for the hearing record, please find enclosed the NRC staff responses.

If the Commission can be of further service, please do not hesitate to contact me.

Sincerely,

Carlton Kammerer, Director
Office of Congressional Affairs

Enclosure:
As stated

QUESTION 1. Mr. Minogue testified that the response of nonrigid piping systems to seismic hazards is being evaluated. Does NRC have a tentative schedule for assessing potential regulatory modifications as a result of this work?

ANSWER.

NRC has a research program to evaluate stiff versus flexible piping to adress the concerns that have been raised that the stiff system may have actually reduced overall safety because it reduces the ability of the system to withstand thermal loads. The presently planned program is expected to be completed in FY 1984. This research will be supported by two other projects. The first project will evaluate the existing design criteria for piping systems with multiple supports that can experience independent motions from an earthquake. This effort will be completed in FY 1983. The second project will evaluate the energy absorption capability of piping systems (damping values). This project will be completed in FY 1984. It is anticipated that appropriate recommendations for modifications to the regulatory design criteria will be made in FY 1985 based on the research results. The most effective way of implementing the recommended improvements in requirements may well be to include them in revisions to the ASME Boiler and Pressure Vessel Code. We will be working to this end through our representatives on the various technical and management committees of the ASME.

QUESTION 2. You attached to your testimony a list of seven examples of the potential future use of NRC research results in the development of regulations. In addition to seismic design of piping and severe accident research, this list included qualification of safety equipment, fire protection, decommissioning, earthquake intensity and disposal of high level waste. Would you tell us when regulations might be modified as a result of research in each of the areas?

ANSWER.

- Seismic design of piping - Research to assess the design criteria for multiple independent supports under earthquake conditions will be completed in FY 1983. Modifications to the Standard Review Plan will be recommended based on the research results. Recommended modifications to the regulatory design criteria for pipe damping and support stiffness will be made in FY 1985 if justified by the research to be completed in FY 1984. This work will also be relevant to the revision of Appendix A to Part 100 discussed later.

- Severe accident research - A "Nuclear Plant Severe Accident Research Plan," NUREG-0900 (DRAFT) has been prepared by the NRC Office of Nuclear Regulatory Research to focus the research on regulatory issues associated with severe accidents. It is intended to provide within two years a technical basis for regulatory decisions concerning design features and operating guides and procedures for existing plants with respect to their ability to prevent and mitigate severe accidents. Following this a program to complete and confirm the bases for regulatory decisions and guidance is projected to take an additional two years.

- Qualfication of safety equipment - Development of the regulatory basis for qualification is separated into two areas - electrical and mechanical - principally because the failure mechanisms are quite different for these two types of equipment (e.g., electrical equipment tends to be more sensitive to temperature and radiation of the reactor environment while mechanical equipment is more sensitive to wear; both can be sensitive to shock and vibration). A new rule on Environmental Qualification of Electric Equipment Important to Safety is in the final stages of NRC review and is to be submitted to the Commission for approval later this summer. It will codify the environmental requirements for qualifying certain electrical equipment important to safety. This rule does not address dynamic qualification. Ongoing research on electrical and mechanical equipment dynamic or seismic qualification will provide procedures for equipment qualification including guidance on practical, justified methods of implementation. These results will be implemented starting in FY 1984. Decisions will be made regarding the most appropriate means of implementation; e.g., regulations, regulatory guides, changes to standard review plan, new or revised national standards, or several of these in combination.

- Fire protection - A new Section 50.48 and Appendix R to 10 CFR Part 50, "Fire Protection Requirements for Nuclear Power Plants Licensed Prior to January 1, 1979," based in part on results of cable separation and flammability fire tests at Sandia National Laboratories under NRC research sponsorship, became effective February 17, 1981. It has been proposed that a comprehensive fire protection rule for nuclear power plants be developed. Fire protection rules will incorporate results from the Browns Ferry Nuclear Plant Fire Replication test and full-scale 20-foot separation tests conducted by Underwriters Laboratory in 1981 and 1982, respectively, for the NRC Office of Nuclear Regulatory Research and from other research tests related to establishing definitive criteria for cable separation distance in different nuclear plant room configurations to be run in FY 1983/84 at Sandia National Laboratories under NRC sponsorship.

- Decommissioning - Research in the decommissioning area is being done to obtain information on status of technology, costs, health effects, and standards for decommissioning a range of nuclear facilities such as pressurized water reactors, boiling water reactors, and various fuel cycle facilities. Based in part on the results of this research, a Draft Generic Environmental Impact Statement (DGEIS) on Decommissioning Nuclear Facilities was issued in January 1981, and a Final GEIS and proposed rules are in preparation for issue in 1983, with effective rules expected to be published about FY 1984.

- Earthquake intensity - The seismic and geologic (earthquake intensity) research is scheduled to result in a revision to Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," of Part 100, "Reactor Site Criteria," (Title 10 CFR) effective in 1988. An accelerated schedule for publication in 1985 is under consideration (by RES and NRR staff).

- Disposal of high-level waste - The research on disposal of high-level waste has been used in developing 10 CFR Part 60, "Disposal of High-Level Waste in Geologic Repositories," now in the last stages of staff preparation for Commission consideration. The research will also be necessary to develop guidance for implementation of the rule.

QUESTION 3. We were pleased to note from Mr. Minogue's testimony that NRC has been moving to reduce the emphasis on narrowly-defined confirmatory research and to deal more directly with broad safety issues. In what ways is the Commission encouraging this trend?

Answer.

The shift in research emphasis away from confirmatory research occurred primarily because of changing Commission needs; however, we feel confirmatory research is still necessary. The Three Mile Island Accident demonstrated the need for a broad and realistic understanding of systems interactions and accident-related phenomena.

The Commission guidance related to this shift stated that Research should, "assure the agency resources are being properly directed toward areas of importance to the licensing and inspection process."

These areas of importance are generally aimed at improving the ability to evaluate the performance of both "safety-related" and "non-safety related" systems and to reduce the uncertainties associated with risk assessment. To adequately understand phenomena and the interaction of plant systems during transients and accidents, a broad systems approach is necessary. This course has been encouraged by the Commission in its review of the:

- research budget;
- Long Range Research Plan; and
- assignment of priorities to Unresolved Safety Issues, Generic Issues, TMI action items;

as well as the requirement to develop a plan for resolution of these identified issues.

QUESTION 4. You mentioned in your testimony a need for continued confirmatory research but assert a better appreciation for the full range of research tasks the agency should undertake. Do you feel that a clear distinction can be made such that each NRC program can be categorized as one or the other?

ANSWER.

Work carried out by the Office of Nuclear Regulatory Research can usually be clearly categorized at the very low level of a task/subtask as being confirmatory or other. However, this distinction within our current structure of defining program objectives has little meaning and is not the basis of justification for the execution of the research.

QUESTION 5. What portions of the regulations have been considered in "analyses of the regulatory structure and safety evaluation practices" under the Systems and Reliability Analysis decision unit? Do you expect this research to expand in the next year or two?

ANSWER.

Analyses of regulatory structure and safety evaluation practices have been undertaken in a number of areas. In safety evaluation practices, the risk significance of the various sections of the Standard Review Plan for reactor licensing was assessed. The Standard Review Plan has been substantially revised, and efforts in this field are underway for the revised Standard Review Plan and the General Design Criteria for reactors.

A substantial amount of regulatory analysis has been performed for reactor siting and severe accident regulation. Preparation of a new siting regulation was recently postponed until 1983 to take advantage of new accident source term data which should be available then. The regulatory analysis for severe accident regulation is woven into our Severe Accident Research Plan and will expand in the next year or two as the technical basis for Commission policy decisions is developed.

In addition, regulatory analyses continue on many other parts of the regulations as part of the periodic and systematic review.

QUESTION 6. Will costs and benefits of research programs be delineated in future editions of the Long Range Research Plan?

ANSWER.

Yes, although there is no explicit cost/benefit criterion for individual research programs. This is one of a variety of factors which were considered but not clearly delineated in the current plan. These include risk significance, regulatory significance, and cost-effectiveness, which will be addressed for major programs in next year's plan.

QUESTION 7. We agree with your statement that the research and user offices must work together to set research priorities. However, isn't one office responsible for justifying an initial position and maintaining the logical integrity of the plan during the comment process? If so, shouldn't this office provide and maintain a systems overview of the research program?

ANSWER.

One office, the Office of Nuclear Regulatory Research, maintains the logical integrity of the research plan during the comment process, and maintains a system overview of the research program. The justification for the initial position is provided by the sponsoring office, which may be Research, NRR, etc. This is compatible with the statement that the responsibility for setting research priorities must be shared by the user offices and the office responsible for conducting the research.

QUESTION 8. In answer to a question about the comparatively limited size of the advanced reactor decision element, Mr. Minogue remarked that the NRC research approach to CRBR is in part dictated by a feeling that NRC should give more attention to what an applicant should do to support his own application. Is this approach considered appropriate for future LWR safety research as well? Will it be applied if post-CRBR LMFBR's or new commercial HTGR's are proposed by the industry?

ANSWER.

Yes, the NRC approach to the CRBR application and possible high-level waste repository applications, that all information necessary to support the application must be supplied by the applicant, will be extended to post-CRBR LMFBR's, new HTGR's, and new generation LWR's.

We are currently moving in the direction of the NRC doing less of the safety research needed for LWR's. The growth of the nuclear industry was very much in parallel with the technology development. A need existed to rapidly develop a regulatory structure supported by a sound technical base. To fill this need in the past, the AEC, and then the NRC, assumed the lead in reactor safety research with facilities such as LOFT and Semiscale which could do generic experiments with broadly applicable data. Much of the need for the very expensive loss of coolant accident LWR safety research developed after a number of LWR plants were already operating, and many were under construction. This required expensive tests to confirm existing systems. In contrast, CRBR and High-Level Waste are single applicant situations and the state of the technology is more advanced.

QUESTION 9. You state in your testimony an expectation that most industry research will be directed towards "economically oriented issues." Do you consider reactor safety research an "economically oriented issue?" Is there any threat to the public health and safety which would result in economic penalities to the licensee?

ANSWER.

NRC's reactor safety research is intended to provide information on which to base regulatory decisions. Once made, these decisions have real economic impact on the public through their application to licensees. From the industry perspective, the issue of economic need arises only after the regulatory decision is made at which time the cost of compliance provides incentive for further research, particularly into cost-effective ways of meeting the regulatory requirement. Industry does conduct research which leads to increased safety on its own initiative. Although much of this work has important safety implications, a primary purpose is to increase plant reliability and efficiency for a greater return on investment. Research programs required to meet the goals of safety and reliability are often congruent; this serves as a basis for much of the NRC/industry coordination in research efforts.

Usually, any threat to the public health and safety arising at a particular licensee facility will have negative economic factors associated with it. These can range from an adverse effect on utility borrowing costs to significant actual costs from plant contamination as occurred at Three Mile Island. However, the perceived cost-risk may represent an insufficient incentive to industry to sponsor safety research in areas where there is a perception of a low probability of occurrence.

QUESTION 10. Were EPRI comments on the Long Range Research Plan (LRRP) publicized in the same way that ACRS comments were? Would you supply a copy of their comments for the record?

ANSWER.

Program Planning between EPRI and NRC nuclear research programs is very closely coordinated on a continuing basis. EPRI's comments on the FY 1983-1987 Long Range Research Plan (LRRP) are attached and were incorporated into the FY 1984-1988 LRRP. EPRI did receive a March 15, 1982 draft of the NRC FY 1984-1988 LRRP, and we have received an early EPRI draft review of our program which we are using internally.

ELECTRIC POWER RESEARCH INSTITUTE

EPRI

December 2, 1981

Mr. Robert Minogue
Director of Research
U.S. Nuclear Regulatory Commission
7915 Eastern Avenue
Silver Spring, Maryland 20910

Dear Bob:

Enclosed is our review, as of September 1, 1981, of the RSR Budget Plan for FY83-84.

Please note that our recommendation on LOFT is consistent with maintaining its general availability without a defined experimental program for the near term.

We are prepared to respond further on any specific questions that the document may stimulate.

Sincerely,

W. Loewenstein for
John J. Taylor, Director
Nuclear Power Division

JJT:WBL:av

Enclosure

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Review of USNRC FY 1983-1984
Internal Review Budget Plan

Edited and Compiled
by

B. K.-H. Sun
W. B. Loewenstein

Nuclear Power Division
Electric Power Research Institute

September 1, 1981

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1. ACKNOWLEDGEMENT

The review and comments documented in this report were performed by the EPRI staff. The following is a list of names, in alphabetical order, who contributed to the effort:

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2. SUMMARY

The NRC released its FY 1983-1984 Internal Review Budget in May 1981. The budget plan shows the priorities and the distribution of NRC research resources in various areas which the NRC perceives as important. It also serves to indicate the licensing concerns which will impact on the commercial nuclear power for the years to come.

The NRC Internal Review Budget is catagorized into nine decision units. The budget of the nine decision units for FY 1981 is shown in Table 1. For each Decision Unit, there are a number of Planned Accomplishments.

TABLE 1
NUCLEAR REGULATORY RESEARCH BUDGET
(DOLLARS IN MILLIONS)

DECISION UNIT	FY81	FY82	FY83		FY84	
	CURRENT	(CONG.)	EDO* REC.	SET ASIDE	EDO REC.	SET ASIDE
LOCA & TRANSIENT	$45.9	$35.1	$31.0	-	$30.3	-
LOFT	42.4	44.0	40.6	-	14.1	-
ACCIDENT EVALUATION & MITIGATION	23.6	35.6	50.4	-	59.3	-
ADVANCED REACTORS	13.1	2.0	2.5	$21.0	2.5	$24.0
REACTOR & FACILITY ENG.	27.8	34.2	38.2	-	40.8	-
FACILITY OPER. & SAFEGUARDS	13.3	16.4	16.8	-	19.1	-
WASTE MANAGEMENT ✓	10.4	22.1	19.6	-	19.6	-
SITING & ENVIRONMENT	12.4	15.3	14.7	-	14.9	-
SYSTEMS & RELIABILITY ANALYSIS	12.8	16.8	21.7	2.0	25.3	2.0
TOTAL PROG. SUPPORT	$201.7	$221.5	$235.5	$23.0	$225.9	$26.0
EQUIPMENT	8.2	7.4	7.9	1.0	7.5	1.2
TRAVEL	0.7	0.9	0.9	-	0.9	-
TOTAL RES	$210.6	$229.8	$244.3	$24.0	$234.3	$27.2

*EDO stands for Executive Director's Office

In the study, each one of the planned Accomplishments is reviewed. The approach of the review is to provide a summary and comments of the NRC budget plan first. The next is to discuss the EPRI projects that are related to the planned NRC accomplishment. This is followed by the identification of areas for possible EPRI and NRC cooperation. Finally, the review addresses the specific recommendations and justifications for the recommendations.

The objective of the study is to assess the NRC research plan in light of possible cooperations between the NRC-RES and the EPRI Nuclear Power Division in the research of nuclear power technology. It provides an evaluation that may be used in the management meetings between NRC-RES and EPRI Nuclear Power Division for information exchange.

3. Major EPRI Recommendations

3.1 LOCA and Transient

(1) EPRI could work with NRC-RES to cosponsor programs to study experimentally the behavior of B&W's one-through steam generators and vent valves during small break and large break LOCAs and transients. The study may include: (a) the Westinghouse FLECHT-SEASET facility for large LOCA study, (b) the B&W BBR facility for small-break LOCA study, and (c) the SEMISCALE facility for system LOCA and transient study.

(2) EPRI should work with NRC to negotiate with GE to participate and cofund the BWR FIST phase II study which are needed for understanding the behaviors of transient and small and intermediate break LOCAs.

(3) EPRI should encourage NRC-RES to use the result of steam generator studies in their modeling efforts and cosponsor with EPRI large testing projects such as the Westinghouse MB-2 tests beyond the current workscope.

(4) EPRI should encourage NRC to initiate or cofund large scale testing to evaluate thermal mixing and thermal resistance during PWR high pressure injection.

(5) NRC should consider restricting the involvement in UPTF (Upper Plenum Test Facility) in Germany beyond 1984.

(6) NRC should consider performing smaller-scale UPTF tests in the U.S. if further data are felt to be necessary.

(7) EPRI should collaborate on 2D/3D data analysis to ensure maximal use of data.

3.2 LOFT

(1) EPRI should work with RES to keep LOFT available as a research facility for nuclear safety and licensing.

(2) EPRI could recommend NRC-RES that there is potential for using LOFT for the following:

a. Cooling of degraded core with fuel rod ballooning and ruptures

b. Operation of reactor with intermittent excursion through the DNB fuel design limit

c. Transport and retention of fission products through the reactor system.

3.3 Accident Evaluation and Mitigation

(1) It would be desirable for EPRI to be represented on NRC's peer research review group for the activities related to fuel melt.

3.4 System and Reliability Analysis

(1) EPRI could assist in the preparation of the National Reliability Evaluation Program (NREP) guide documents, either by direct effort and/or technical review.

(2) EPRI should participate in the review of the checklists of the hypothetical deficiencies in plant design which NRR plans to deduce from selected PRA studies for use by NRR and IE in licensing and inspection.

3.5 Siting and Environment

None

3.6 Reactor and Facility Engineering

(1) EPRI could have a cooperative effort with NRC in the areas of
(a) surveillance techniques for on-going equipment qualification and
(b) quantification of cost and benefit based on overall risk assessment.

(2) NRC criteria development efforts for decommissioning should be followed by EPRI to be sure that utility interests are at least considered, and that needs for technical support of site disposition decisions and actions are identified well in advance.

3.7 Waste Management

None

3.8 Facility Operations and Safeguard

(1) A systematic R&D report exchange program in the Human Engineering area should be established. At the present time, this exchange is rather informal in nature.

(2) The NRC program should account for the improved assessment of the fission product source term in the accident scenarios and the requirements of the emergency planning rule.

(3) NRC R&D appears committed to an assessment of the Disturbance Analysis and Surveillance Systems (DASS) in FY1984. EPRI should collaborate in ensuring adequate review and assessment.

(4) The DASS effort provides an integrated technical framework for coordinating I&C and computerized operator aids. NRC involvement would assist future development.

(5) NRC should give higher priority and consideration to the technical bases for emergency preparedness requirements, on the needs of the licensed activities and on the interfaces between licensees and federal and state agencies. NRC also needs to clarify the distinction between preaccident emergency preparedness requirements and those requirements that would be operative for the actual management of a real emergency.

(6) NRC should examine its posture on the development of accident scenarios for exercising the emergency plans, particularly from the standpoint of Commission policy for defining the appropriate emergency drills.

3.9 Advanced Reactors

(1) EPRI should be cognizant of the HCDA related work and its findings relative to licensing of LMFBR in the future. The relationship of the work pursued to the IDCOR work should be explored.

(2) EPRI must maintain at least a minimum credible R&D program in Fast Reactor technology in order to have a substantial input and perhaps influence in the dialogue with NRC.

4. SPECIFIC REVIEWS AND COMMENTS

4.1 LOCA and TRANSIENT

The LOCA and Transient Research used to be the largest research area in NRC-RES programs. The projects in this area are being significantly trimmed down, redirected, or phased out after the TMI-2 accident. The planned accomplishments and their associated budget are indicated below:

LOCA and Transient Budget 1981 - 1984 ($M)

Planned Accomplishment	1981	1982	1983	1984
1. SEMISCALE	7.8	7.5	7.5	7.5
2. Separate Effect Experiments and Model development	11.2	6.7	6.6	6.1
3. 2D/3D Reflood Program	9.4	6.0	6.5	7.0
4. Code Improvement & Maintenance	5.9	3.8	2.6	2.1
5. Code Assessment & Application	4.2	6.0	6.9	7.1
6. Fuel Behavior Under Operational Transients	7.8	5.1	0.9	0.5
TOTAL	46.3	35.1	31.0	30.3

4.1.1 SEMISCALE and SEPARATE EFFECTS EXPERIMENTS AND MODEL DEVELOPMENT

4.1.1.1 Summary and Comments

The SEMISCALE Program is the work horse of NRC-RES to produce thermal-hydraulic data for phenomenological understanding, code development, and code assessment in the areas of PWR plant transient, small and large break LOCAs. Specific testing includes large break LOCA with steam generator tube rupture, small break LOCA with operator actions, and natural circulations with and without noncondensible gases. The SEMISCALE program also includes the development and the assessment of the RELAP 5 code which is users-oriented and designed for LWR applications, in contrast to the complicated scientific TRAC code.

The planned accomplishment of Separate Effects Experiments and Model Development is unique in the NRC-RES plan because (1) the major programs are carried out by reactor vendors, i.e., General Electric and Westinghouse, and (2) the major programs are co-sponsored with EPRI and the vendors. The work-scope includes system behavior under natural circulation conditions and the performance of ECCS under large and small break LOCA conditions for both PWRs and BWRS.

4.1.1.2 Relationship to EPRI Programs and Projects

Beside keeping track of the progress of the SEMISCALE program, EPRI's participation has been in the evaluation of selected SEMISCALE data for NRC Standard Problems (RP695).

In the Separate Effects Experiment and Model Development area, EPRI and NRC cosponsor three major projects. They are (1) PWR Reflooding and Natural Circulation Tests, or FLECHT-SEASET (RP959), (2) BWR Refill and Reflood Tests (RP1377), and (3) BWR Full Integral System Test (RP495).

4.1.1.3 Areas for Possible EPRI and NRC Cooperation

(1) EPRI could perform pretest and post-test analyses for selected SEMISCALE tests to assess the RETRAN and other EPRI codes.

(2) EPRI could suggest to NRC test conditions for severe LOCA tests, such as LOCA with Steam generator tube failures. This would mean EPRI participation in review group meetings.

(3) EPRI could consider cofunding with NRC the limited B&W program to evaluate the performance of vent valves in LOCAs and operational transients.

(4) EPRI should continue to co-sponsor programs related to the PWR and BWR ECCS performance such as PWR FLECHT-SEASET, BWR Refill-Reflood, and BWR FIST.

4.1.1.4 Specific Recommendations and Justifications for the Recommendations

(1) The behavior of B&W's one-through steam generators and vent valves during small break and large break LOCAs and transients have not been studied experimentally. EPRI could work with NRC-RES to cosponsor programs in this area. The study may include the Westinghouse FLECHT-SEASET facility for large LOCA study, the B&W BBR facility for small break LOCA study, and the SEMISCALE facility for system LOCA and transient study.

(2) EPRI should work with NRC to negotiate with GE to participate and cofund the BWR FIST phase II study. The BWR FIST facility which is currently under design and construction is the only BWR facility capable for transient and LOCA studies. The data are needed for understanding the behaviors of transient and small and intermediate break LOCAs.

(3) While many of the reactor transients are initiated from the secondary loop, the performance of the steam generators during such transients or LOCA have not been studies in detail by NRC-RES. Having substantial work done in this area, EPRI should encourage NRC to use the results and cosponsor with EPRI large testing projects such as the Westinghouse MB-2 tests beyond the current EPRI program.

(4) The PWR thermal shock problem is one of the active licensing issues to date. The 1983 - 1984 NRC budget does not identify it as a subject of thermal-hydraulic research. The current EPRI authorization for R&D in the thermal-hydraulic thermal shock area only warrants limited studies. EPRI should encourage NRC to initiate or cofund large-scale testing evaluate thermal mixing and thermal resistance during high pressure injection.

4.1.2 2D/3D PROGRAM

4.1.2.1 Summary and Comments

This is the giant three-party reflood program with

(1) Japanese (JAERI) Cylindrical 2000 rod core (CCTF) facility

(2) Japanese (JAERI) Slab Core (SCTF) facility

(3) German (BMFT) PKL Facility - core 2

(4) German (BMFT Upper Plenum Hot leg injection tests (UPTF)

(5) U.S. (NRC) TRAC Analysis for Support to (1) - (4)

(6) U.S. (NRC) Instrumentation Supply to (1)-(4)

Although not clear in the write-up, this is a 2-300M$ program subject to International Agreements. Note that the U.S. has no experiments.

The Japanese tests are the largest to date, and are either complete (CCTF) or underway (SCTF). The German UPTF was scheduled for completion by 1985 and the Germans are requesting extension. This appears to be contrary to the original (signed) agreement. The key questions are:

(1) Data availability - it is difficult to obtain and publish due to Japan licensing and KWU commercial concerns. It is a significant question whether this foreign data can be satisfactorily qualified for U.S. application.

(2) Program control - U.S. industry has only a minor role in the program. Some review groups have been held with EPRI attendance. Only recently (June '81) has NRC asked for detailed EPRI review of UPTF due to cost escalation and schedule slippage (now 1987).

(3) The U.S. commitment to UPTF is questionable since it supports a non-U.S. reactor design, manufacturer and safety case (the German KWU hot leg injection).

(4) No U.S. experiments are being built as part of this program; and instrumentation supply to on-going U.S. programs is limited as a consequence of this foreign commitment.

4.1.2.2 Relationship to EPRI Programs and Projects

The projects are closely related to Enginering Safety Features performance, small break and large LOCA studies.

(1) RP959 (jointly with NRC/W) FLECHT-SEASET Supplied core blockage definition to SCTF tests; RP1378 Supplies core blockage correlations.

(2) RP1731 studies small break behavior in loops, as in recent PKL tests. EPRI has just received the PKL report directly through our exchange agreement with BMFT (NB not through NRC).

(3) RP1159-4 has developed two-phase void and flow regime instrumentation (the "Auburn" devices). NRC only uses instruments from its own R&D program at national laboratories (INEL and ORNL) - EPRI would like to see their instruments applied in large tests.

(4) RP1377 Refill/Reflood examines upper plenum phenomena and core interactions during water injection, at a large GE testing site (Lynn).

(5) One task of RP1118 was specifically designed to study multidimensional flows in a 2000 pin small scale (1/6 height) slab model of SCTF. NRC is very interested in their study, which also involves ACRS consultants as investigators.

4.1.2.3 Areas for Possible NRC and EPRI Cooperation

(1) Perform U.S. UPTF test in Lynn facility concentrating on U.S. design.

(2) Use of EPRI instrumentation from 1159-4 on 2D/3D program.

(3) Data analysis from CCTF and SCTF.

4.1.2.4 Specific Recommendations and Justifications

(1) The UPTF in Germany should be deleted from the program beyond 1984. The rationale for this is the results are (a) not timely for U.S. program due to delays, (b) significant cost escalation, (c) a non-U.S. design PWR will be tested, and (d) formally the German request changes the 3-way agreement, and U.S. is not obliged to commit further to the program.

(2) Performing smaller-scale UPTF tests in U.S. is a viable option. These could be used for trading for UPTF results later. The U.S. would gain (a) control of program, both in timescale and budget, (b) a U.S. design could be tested, perhaps with vendor participation, (c) results could contribute to U.S. rulemaking scheduled for 1984, and (d) earlier results (1983-5) are obtained

(3) Ensure degraded ESF tests are included is an important objective to:
(a) define margins in current assumptions (single failure, etc.),
(b) give timescales for operator actions and for procedure evaluation,
(c) remove large LOCA emphasis and place program into modern safety perspective

(4) EPRI should collaborate on data analysis in order to ensure, (a) maximal U.S. use of data other than to show "conservatisms" within NRC and (b) test whether one dimensional codes are adequate, or can be correlated with these data.

4.1.3 CODE IMPROVEMENT and MAINTENANCE, and, CODE ASSESSMENT and APPLICATION

4.1.3.1 Summary and Comments

These planned accomplishments appear to be leading to codes which are too complicated to be useful. They stress (a) multidimensionality of assessment of operator error and equipment malfunctions and (b) B.O.P. modeling and imply the capability to benchmark against data. The following are comments:

(1) These codes will run very slowly hence will not be useful for the implied need ("rapid and accurate response to NRR requests for licensing assistance").

(2) The experimental data needed to produce the closure relations for the -implied- six equation 3D models do not exist (they do mostly exist for 5 eq. 1-D models).

(3) The experimental data for multidimensional hydrogen transport in containment does not exist to validate the NRC proposed codes (although EPRI is developing certain aspects of that data base).

(4) The COBRA code (as well as RELAP) is physically insufficiently based to produce reasonably accurate results for BWR transients.

(5) The engineering analyzer appears to be poorly conceived since each plant is sufficiently different to make it unlikely to be useful for its proposed purpose (see the DOE report on the industry consensus).

4.1.3.2 Relationship to EPRI Programs

EPRI is working on many of these items but their end uses are not perceived to be the same hence some of the limitations do not apply:

(1) EPRI is working on BOP modeling and simultaneously trying to gather existing data on BOP components.

(2) EPRI is not attempting to produce detailed 3D models to run in-line with full system representations.

(3) EPRI is looking towards the development of a multicompartment (but not multidimensional) containment code for H_2 evaluation and are developing part of the data needed to evaluate and verify such a code.

(4) EPRI is examining advanced system modeling concepts to remove already perceived limitations in RETRAN.

(5) EPRI is developing VIPRE (an advanced form of COBRA) which will be carefully benchmarked but will still have some problems relative to 3D closure relations.

4.1.3.3 Areas of Possible EPRI and NRC Cooperations

Cooperation in the codes area appears to be most likely to succeed if it is in the physics area.

The codes RETRAN/RELAP 5; VIPRE/COBRA, etc., while meant to ultimately solve the same problems use seriously different approaches to the solutions. It does not appear then to be feasible to expect a cooperative effort in specific code development to be successful. The physics portion of the codes however comprise mainly

1. The number of thermal hydraulic equations (4, 5, 6, etc.)
2. The constitutive (closure) relations
3. Heat transfer coefficients
4. Friction factors

It would appear to be possible to decide on the class of phenomena expected to be modeled (hence to determine the number of equations) and to set up projects which would by analyzing extant data (and perhaps recommending further experiments) determine "Best Fits" for the needed closure relations. Such Best Fits would include:

1. Slip correlations

2. A continuous boiling curve (i.e., heat transfer coefficients)

3. Continuous friction factors for:

    a. wall to liquid

    b. wall to vapor

    c. liquid to vapor

4. Bulk vapor condensation coefficients

5. Bulk liquid superheating

6. Vapor superheating in the presence of liquid

There are undoubtedly others which could be listed but need not be at the present time. The use of maps which are discontinuous and which lead to serious numerical problems should be excluded.

4.1.3.4 Specific Recommendations

If interest exists a NRC/EPRI technical group should be established to work out the details.

4.1.4 Fuel Behavior Under Operational Transients

4.1.4.1 Summary And Comments

The title and thrust are somewhat misleading based on the history of this funding area, but is descriptive in some sense of the 1983 and 1984 efforts. The FY1983 and FY1984 research in this area deal primarily with the FRAP-T (transient fuel behavior code) and FRAPCON (steady state fuel behavior code) computer codes. Maintenance of FRAP-T, Fuel code assessment and special assessment of fuel codes with commercial fuel rod data (EPRI's RIMS data base) efforts are included.

In FY1981 and FY1982 (and prior years) this category (or a somewhat differently named category) contained significant experimental efforts in the PBF, NRU, and Halden reactors, and a number of ex-pile experimental programs at Argonne National Laboratory, ORNL and others on the properties of zircaloy. It appears that the ORNL multirod burst test (MRBT) program, FRAPCON model developments and other efforts has been shifted from here to "Accident Evaluation and Mitigation" decision unit under "Behavior of Damaged Fuel" beginning in 1983.

The major thrust of FRAPCON has been the prediction of transient/accident initial conditions. The major thrust of FRAP-T has been to predict fuel behavior during transient/accident scenarios. The initial FRAP-T thrust was the prediction of LOCA and other transients with recent funding justification words emphasizing "operational transients." The semantics aside, the core direction of code development has not changed from one of producing a "best estimate" tool to look at a spectrum of accidents with LOCA as the most severe.

This effort has always been segregated from the thermal hydraulics/heat tansfer efforts and has suffered as a result. The heat transfer package is the same as in RELAP with added CHF correlations. Only limited effort has been made to enhance or match the heat transfer package to the fuel rod thermal mechanical prediction package, and to the material properties package, all which have been funded since 1973.

4.1.4.2 Relationship to EPRI Programs and Projects

EPRI's transient fuel behavior program is contained in RP1117, RP1321 and RP1627. The current active components of this program are the FREY code development (RP1321-4) which is similair to FRAP-T in that it is intended for realistic analysis to transients. EPRI is organizing a data base (ERUDITE) which will be used to assess FREY. It consists primarily of U.S. Government sponsored test data from PBF, SPERT, and some single rod tests for MRBT.

EPRI's work on the COMETHE and SPEAR codes is similar to NRC's FRAPCON, in that they are all intended for realistic analysis of steady state operation of fuel rods in nuclear power plants.

4.1.4.3 Areas for possible EPRI and NRC Cooperation

EPRI requested in 1980 access to the MRBT data, specifically requesting data tapes for inclusion in ERUDITE. EPRI understands that the bundle data have not been analyzed or documented to the point that it is releasable in magnetic tape (this includes tests completed Sept. 1977, Nov. 1977, and Aug. 1978). Limited distribution of ORNL/NUREG's were issued in 1979 for the 1977 tests (ORNL/NUREG/TM 322 and ORNL/NUREG/TM 337). EPRI would like to obtain copies of these reports.

However, it is EPRI's understanding based on discussions with knowledgable individuals, that the reports are inadequate for our purpose, which is inclusion in the ERUDITE data base. EPRI would like to obtain the analyzed data tapes already requested. It is our understanding the NRC would have to fund ORNL to get the data into "usable form." In any case, it is more cost efficient to EPRI to handle data tapes rather than utilize data from data reports by hand. A cooperative effort in this area is clearly desirable.

NRC Research has not attempted to QA the data with which they qualify and verify their fuel codes. NRC has relied on large statistical samples of data for comparison to insure they have no problems. EPRI has chosen a more selective data choice and is attempting to QA the transient fuel behavior data being put into ERUDITE. NRC's continued cooperation in providing PBF data tapes would be appreciated. Encouragement to provide other data tapes in the future from other NRC programs (such as the one currently running in NRU on LOCA reflood behavior) will be beneficial to both EPRI and NRC. NRC's cooperation in providing data tapes, that can be easily used, would help control EPRI costs in including the data in ERUDITE. NRC contractor QA work in support of EPRI QA activities may be an important area of cooperation. NRC cooperation and assistance in QA activities would not only assure the quality of the data for use in EPRI's data base, but also would help insure quality data for NRC's use.

4.2 LOFT (LOSS OF FLUID TEST)

4.2.1 Summary and Comments

The NRC budget for the LOFT program as shown in the table indicates clearly its plan to terminate testing at the end of fiscal year 1983.

LOFT 1981 - 1984 Budget ($M)

Planned Accomplishment	1981	1982	1983	1984
(a) tests and operations	28.6	29.1	20.1	0
(b) analysis and reporting	9.7	10.4	8.5	2.2
(c) fuel	4.1	4.5	1.9	1.2
(d) program close out	0	0	5.4	1.1
(e) standby activity	0	0	4.7	9.6
all of LOFT	42.4	44.0	40.6	14.1

There are three tests to be conducted in 1983; one small break LOCA, one anticipated transient without scram (ATWS), and the last one to test large break LOCA with fuel rod ballooning and rupture. The reporting will include the usual quick look report, data report, pre test prediction report, and post-test analysis report. The analysis work is to evaluate LOFT data with respect to the calculated responses of commercial PWRs and to present data in a manner that facilitates NRC confirmation to nuclear reactor analytical methods. Destructive fuel examination will be performed pertaining to NRC licensing.

The staff rundown and severance will begin in 1983. The facility will be maintained in a standby mode for two years, during which operating reactor experience and plans for possible reactivation of testing will be examined.

The NRC plan for LOFT basically complies with the recommendations made by the Special LOFT Review Committee in Feb. 1981. It is different from the recommendations of the ACRS who has been persistently arguing to terminate LOFT by the end of FY1982.

4.2.2 Relationship to EPRI Programs and Projects

The EPRI staff and projects have interacted with the LOFT program in the following cases:

(a) EPRI performed pretest and post-test analyses of selected LOFT LOCA and operational transient tests (RP695) . EPRI participated in NRC Standard Problem Evaluation to assess the EPRI RETRAN code against LOFT data (RP695) . EPRI used a LOFT small break LOCA test to check the performance of a water level gage (RP1611)

(b) EPRI has participated in periodic LOFT Review meetings. A LOFT personnel has served in the EPRI DASS advisory group (RP891)

4.2.3 Areas for Possible EPRI and NRC Cooperation

The NRC has formally requested EPRI to co fund the LOFT program (letter from Dircks to Culler, dated Nov. 12'81). An EPRI staff proposal to fund $2M per year was deferred to beyond CY86. The action precludes near term formal participation of EPRI in the LOFT Program management and direction.

Since the LOFT program now is under close NRR scrutiny as a result of the Special LOFT Review Group recommendations, the future LOFT data are expected to be used for licensing decision making. An example of this is the pump on/off issue for small break LOCA. NRR decided after evaluation of LOFT pump on and off tests in 1980 and 1981 that PWR operators should shut off recirculation pumps in the event of small break LOCA.

EPRI should continue to participate in the analysis of LOFT data, program review, and making suggestions to NRC about future LOFT directions.

4.2.4 Specific Recommendations and Justifications for the Recommendations

(1) Although there are scaling compromises, the LOFT facility remains the largest experimental PWR with advanced instrumentation. Without LOFT, it is possible that individual utilities will be requested to use their reactors as test beds to provide plant performance data and to demonstrate safety. Therefore, it may be beneficial to industry to have LOFT available to produce generic data for addressing safety licensing issues. In fact, in 1980 ACRS strongly recommended that LOFT should be funded by the industry.

EPRI should work with NRC-RES to keep LOFT available as a research facility for nuclear safety and licensing. It is our opinion that LOFT can be operated at a budget level of about $20M to $25M per year with emphasis on testing.

(2) The mission of LOFT has been changed several times since 1962 from the study of core melt to demonstration of large LOCA safety to the current code assessment. As the concern of degraded core cooling and fission product release becomes the central licensing issue, the LOFT facility may play a major role in providing data for such licensing evaluations. EPRI could recommend NRC RES that there is potential for using LOFT for the following:

   (a) Cooling of degraded core with fuel rod ballooning and ruptures.

   (b) Operation of reactor with intermittent excursions through the DNB fuel design limit

   (c) Transport and retention of fission through the reactor system

4.3 ACCIDENT EVALUATION AND MITIGATION

The Accident Evaluation and Mitigation decision unit includes four planned Accomplishments. Their budget is listed as follows:

Accident Evaluation and Mitigation Budget 1981-1986 ($M)

Planned Accomplishment	1981	1982	1983	1984
1. Behavior of Degraded Fuel	15.7	17.4	25.7	26.5
2. Fuel Melt	4.0	10.0	13.7	16.1
3. Fission Product Release and Transport	1.9	4.3	5.4	7.9
4. Service Accident Mitigation	2.0	3.9	5.6	8.8
Total	23.6	35.6	50.4	59.3

4.3.1 BEHAVIOR OF DAMAGED FUEL

4.3.1.1 Summary and Comments

Severe accident sequence analysis to give system and coolant conditions leading to severe fuel damage, and characteristics and coolability of debris bed will be studied. In pile experiments in PBF are to give fission product release and hydrogen generations rates. Other experiments will concentrate on coolability of ballooned and burst cladding (both in-pile and ex-pile). Operational PBF costs are included.

An improved version of Severe Core Damage Analysis Package (SCDAP) will be completed for use in Degraded Core Rulemaking.

PBF tests on fuel involve "seasoned" fuel with an irradiation of a few hours. Thus fission product levels are probably tracer or atypical. The "seasoning" may not produce a fuel with typical properties of a 30,000 MWD/T specimen. However, if the NRC can deliver on the whole program it will be very helpful.

The ACRR tests use one foot long rods and cannot give realistic results for a prototypical power plant, but can be useful in studying phenomena.

NCR's participation in ESSOR used to be under LOCA/Fuel Behavior. The test matrix has changed annually as the program has slipped.

4.3.1.2 Relationship to EPRI Projects

RP1233 (SAI) directed at improved model for fission product release from fuel (in part).

RP1431 (UCLA and ANL) working on coolability of hot debris under forced flowing conditions and under pool boiling conditions. It is intended to develop a Primary Coolant System thermal/hydraulics code. This effort in conjunction with RP1933 will generate a code comparable to SCDAP.

E007 will carry out measurements on chemical form of the fission product released.

4.3.1.3 Areas for Possible EPRI and NRC Cooperation

It is desirable that industry (EPRI) and NRC develop independent computer codes for analysis of severely degraded accidents. A reciprocal peer review activity has been established.

4.3.1.4 Specific Recommendations and their Justifications

A program not covered by either NRC or EPRI is in pile loop experiments which degrade the fuel and follow the rate of released FP's in the plumbing of the loop (prototypical of a reactor). PBF work does not go far enough since fuel is not typical; also degraded fuel should also be able to be exposed to steam and hydrogen as well as water.

4.3.2 FUEL MELT

4.3.2.1 Summary and Comments

Experimental and analytical work is included to give data base in severe accident progression from behavior of damaged fuel to source term. Analysis development includes attack of core material on vessel with drop of core into reactor cavity, attack on basemat, containment response. Steam explosion and spike work are part of this program.

Hydrogen combustion is to be studied. Determine debris bed coolability is proposed. (This work appears to be at Sandia although not identified). This work on debris bed coolability seems to duplicate NRC work on "behavior" of damaged fuel. The data from this program when acquired seems pertinent.

The proposed hydrogen work appears to duplicate EPRI's project. NRC may be prepared to review EPRI's project and probably curtail and coordinate their own effort.

4.3.2.2 Relationship to EPRI Projects

RP1933 is developing an improved model for core slumping, attack on vessel and steam spikes. EPRI has no plans for experimental work in these subjects.

RP1931 is supporting experimental work on debris bed and interaction of corium and water (steam spikes).

RP1932 is supporting a comprehensive hydrogen combustion and management program.

4.3.2.3 Specific Recommendations and their Justifications

It would be desirable for EPRI to be represented on NRC's peer research review group for these activities. A coordinated EPRI/NRC effort may then be achieved in this area as discussions proceed and the program plans mature.

4.3.3 FISSION PRODUCT RELEASE AND TRANSPORT

4.3.3.1 Summary and Comments

NRC will release of fission products from damaged fuel, including molten fuel, transport through reactor coolant system and containment; also study release of fission products and nonradioactive aerosols from corium concrete interactions also transport behavior of multi component aerosols. The radiation environment for equipment qualification and functioning of engineering safety features is included. Collect and evaluate data from TMI-2. Extension of TRAP-MELT to containment fission product behavior (seems to be ORNL program) is included.

4.3.3.2 Relationship to EPRI Projects

RP1233 working on source term and developing program plan for aerosol modeling.

RP1931 will study aerosol generation.

RP1933 will model core debris concrete interaction.

RP2120E involves small-scale primary coolant system aerosol work.

E006 will develop a computer code for high concentration aerosols in a hydrogen steam environment and support EPRI participation in a large scale experiment.

4.3.3.3 Areas for Possible EPRI and NRC Cooperation

EPRI is exploring the possibility of large scale aerosol experiments in the Primary Coolant System at Marviken. It has been suggested to NRC as a subject for joint cooperation.

4.3.3.4 Specific Recommendations and their Justifications

It is appropriate for industry (EPRI) and NRC to develop independent computer codes to analyze severely degraded core accidents. A reciprocal peer review activity has been established.

4.3.4 SEVERE ACCIDENT MITIGATION

4.3.4.1 Summary and Comments

Evaluation of new strategies for mitigation of severe accident consequences. Results are to support licensing actions. Evaluations will include filtered vented containment, core retention systems, hydrogen control systems, containment heat removal.

This planned accomplishment supports NRC's approach to degraded core rulemaking.

4.3.4.2 Relationship to EPRI Projects

RP1932 working on hydrogen control systems.

RP1233 and RP2117E work on water pools.

RP1933 involves core, debris concrete interaction modeling.

RP1933E involves containment behavior code development.

E008 and E009 will study structural capacity of containments to internal pressure.

4.3.4.3 Areas for Possible EPRI and NRC Cooperation

As usual there is a real need for communication and cooperation to avoid undesirable duplications.

4.4 SYSTEM AND RELIABILITY ANALYSIS

The system and Reliability Analysis decision unit comprises three planned accomplishments. The tabulation of their budget is shown below:

Budget 1981-1984 ($M)

Planned Accomplishments	1981	1982	1983	1984
(1) Risk Methodology and Data Analysis	5.6	7.5	8.0	9.5
(2) Reactor Reliability and Risk Analysis	5.1	7.0	11.0	13.0
(3) Transportation and Materials Risk Analysis	2.1	2.3	2.7	2.8
Total	12.8	16.8	21.7	25.3

(1) and (2) are roughly equal in effort and both nearly double in real terms over '81 '84. (3) roughly constant effort at $2M/$3M per annum.

General Comments

The NRC Plan will extensively use PRA to evaluate the effectiveness of regulations and standards and to help structure and support rulemakings to improve new regulations and standards. This will have a direct effect on the utility operations.

NCR will publish many reviews, guides, manuals on PRA procedures, methods etc for the use of their own staff as well as utilities. A list of these reports are:

(1) Regulatory Guide and Standard Review Plan for NREP (National Reliability Evaluation Program) submittals

(2) Users Guides for NREP application

(3) Regulatory Guide on failure rate data base for NREP sequence screening

(4) List of features hypothetically leading to dominant risk contributions

(5) Analysis of effectiveness of the General Design Criteria and Standard Review Plan

(6) Preliminary fuel cycle risk assessment

(7) PRA manuals for each element of the fuel cycle

The NRC Methodology concentrates on system dependencies, common cause failures, and human error. Broadly speaking, this seems exactly right. The EPRI reliability subprogram will focus on this area (EPRI Subprogram 1311, Projects E005 Probabilistic Analysis and Application, E010 Common Cause Studies, E014 Safety Related Human Error).

4.4.1 RISK METHODOLOGY AND DATA ANALYSIS

4.4.1.1 Summary and Comments

(1) Concentration on multiple failures and human error is appropriate.

(2) The current NRC Procedures Guide on PRA is heavily based in its treatment of multiple failures towards design intended failures or system interdependence. This is important but relatively tractable. The crux of the common cause failure dilemma is elsewhere, in design faults, test, and maintenance errors and operational procedures. The EPRI reliability subprogram 1311 (E010, E014) will address these issues. The extent to which NRR plan to do so is not clear though a study of human error in some of these areas is mentioned.

(3) The trend to automatic and comprehensive data acquisition should generate a large amount of data. Its value may depend to a large degree on the cooperation of utilities. It should be watched closely since the data are the basis of PRA. The data should be based on a sound engineering appreciation with a minimum of statistical manipulation. Data acquisition and analysis form an important part of our reliability subprogram (1311, 1233-1)

4.4.1.2 Relationship to EPRI Projects

EPRI's reliability subprogram (1311, 1233-1) is establishing a failure data base in the areas of ATWS, diesel generators, loss of off site power pumps, and valves, and this will be extended to establish common cause failure data in a variety of major systems (1311, E010). There may be two ways in which these activities could interact with the NRR data collection activities to avoid duplication and to perform data comparisons to benchmark some data sets and resolve differences, particularly in the common cause data.

4.4.1.3 Areas for Possible EPRI and NRC Cooperation

Failure data is often politically or commercially sensitive information. However, NRC will establish their data base notwithstanding, and EPRI would act in trying to validate it at source instead of after the fact.

4.4.1.4 Specific Recommendations and their Justifications

EPRI could assist in the preparation of the NREP project guide documents, either by direct effort and/or technical reviews.

The justifications are:

(a) NREP will consist of licensee performed risk assessments.

(b) The guides must be technically correct, clear, and easy to use.

(c) The procedures and guides should preferably produce risk assessments that can be used or adapted by utilities themselves for various purposes to be studied in EPRI reliability subprogram (1311, E013) e.g., safety significance of a change in maintenance schedule or to study financial risk or derive feedback to design.

4.4.2 REACTOR RELIABILITY AND RISK ANALYSIS

4.4.2.1 Comments

(1) The commission and various TMI enquiries have urged the increasing but careful use of PRA. The Review Plan shows extensive all around use of PRA and concentrates on

- a. Evaluating regulatory requirements
- b. Comparing alternative design options

EPRI should basically support these aims, being wary of supplanting common sense and engineering judgment with statistics. EPRI subprogram 1311, project E013, Documentation and Utilization of PRA plans to extend utility utilization of PRA in several directions.

(2) Considerable value impact studies will be performed on various risk assessments for different purposes (see Eg. 2.2). If this is the case EPRI needs to know how to draw well structured conclusions from PRA. The best ways may not involve value impact studies for all purposes. The reliability subprogram will address this as part of PRA utilization studies (1311, E013).

(3) Note the checklists to be prepared for NRR and IE to be used in licensing and inspection.

(4) Note the use of PRA and Operations Research to analyze the effectiveness of the General Design Criteria and Standard Review Plan in 1983.

(5) Note the PRA on gas cooled reactors.

4.4.2.2 Specific Recommendations and their Justifications

The NRR plans to deduce the hypothetical deficiencies in plant design from selected PRA studies. EPRI should participate in the review of the checklists for NRR and IE use in licensing and inspection.

The justification for the recommendation is that the results will be used to:

(a) Establish future PRA improvements

(b) Produce checklists for NRR and IE to use in licensing and inspection.

(c) Eliminate blind spots in operations and maintenance.

(d) Improve guides for accident diagnosis.

All of these will have an impact on utility operations and the requirements they must fulfill.

4.4.3 TRANSPORTATION AND MATERIALS RISK ANALYSIS

4.4.3.1 Comments

(1) Note preliminary PRA on the nuclear fuel cycle in '83 plus peer review.

(2) Note preliminary PRA on transport of nuclear fuel in '83 primarily to reassess regulations.

4.5 SITING AND ENVIRONMENT

This Decision Unit consists of four planned accomplishments. Their budget breakdown for the years between 1981 and 1984 is as follows:

Planned Accomplishments	1981	1982	1983	1984
(1) Health Effects	2.9	3.9	3.0	2.8
(2) Earth Sciences	6.2	7.2	7.1	7.1
(3) Siting	1.3	2.0	2.8	3.5
(4) Environmental Impacts	2.0	2.2	1.8	1.5
Total	12.4	15.3	14.7	14.9

4.5.1 HEALTH EFFECTS

4.5.1.1 Summary and Comments

Experimental and analytical support for NRC regulations related to public and worker health. Implement revision to 10 CFR 20. Amend regulations related to nuclear medicine. Examine occupational doses from neutrons. Feasibility study for potential register of radiation workers.

NRC proposes by 1984 to reunite 10 CFR 20 following 1 CRP 26. Existing 3 Rem/quoted limit with cummulative limit of 5 Rem/year likely to be replaced by 5 Rem/year. Permissable to go to 10 Rem/per incident but less than 25 Rem per lifetime. Report annual exposure for all workers. Minimum impact upon utilities expected.

Conventional dosimeters do not respond to low energy neutron spectrum in containments.

4.5.1.2 Relationship to EPRI Projects

The Nuclear Power Division has no related projects. The Occupational Health Subprogram in Energy Analysis and Environment Division has two related projects.

(1) Assess health effect, if any, of exposure to ionizing radiation among utility employees.

    (a) mortality study
    (b) morbidity study

(DOE is studying reproductive effects)

(2) Assess neutron exposures and effects on utility employees.

4.5.2 EARTH SCIENCES

4.5.2.1 Summary and Comments

Program is aimed at establishing reliable data base and analysis method to verify the site safety assessments for nuclear facilities. Research includes sub surface exploration and earth stress measurement in the Eastern and North western U.S., tectonic model development for S.C. and Mo. Seismogenic Zones, seismograph network digital data analysis, liquefaction prediction and volcanic hazards, and meteorological study.

It seems that NRC should coordinate with USGS for some of the efforts if this is not being considered or included.

4.5.2.2 Relationship to EPRI Projects

EPRI does not have any projects in this area.

4.5.2.3 Areas for possible EPRI and NRC Cooperation

EPRI is interested in research results achieved in this program.

4.5.2.4 Specific Recommendations and their Justifications

Although the program's objective is to support domestic licensing and regulatory guide, analysis methodology can also be benefitted from foreign seismological and geological data.

4.5.3 SITING

4.5.3.1 Summary and Comments

Siting activities comprise one element of the NRC's Siting and Environment Branch which employs 30 people and plans to spend $14.7 million in 1983. The siting branch has a $2.8 million budget in 1983.

Continuing research and standards programs dealing with both power plant and fuel cycle facility siting including (1) comparing alternate sites proposed by licensees; (2) obtaining additional data on population density land use patterns and siting alternatives; and (3) providing validated technical bases for standards and siting criteria.

4.5.3.2 Relationship to EPRI Projects

None within Nuclear Power, possibility indirectly related activities in EA&E Division.

4.5.4 ENVIRONMENTAL IMPACTS

4.5.4.1 Summary and Comments

Experimental and analytical support for NRC's Environmental Impact Statements required by NEPA.

4.5.4.2 Relationship to EPRI Projects

EPRI has no projects related to environmental effects of radioactive pollution.

4.6 REACTOR AND FACILITY ENGINEERING

The Reactor and Facility Engineering decision unit comprises six planned accomplishments. Their budgets for the years 1981 to 1984 are indicated below:

Planned Accomplishments	1981	1982	1983	1984
(1) Mechanical and Structural Engineering	9.0	11.4	12.7	14.2
(2) Primary System Integrity	12.7	14.4	16.4	17.5
(3) Electrical Equipment Qualification	2.6	4.0	3.8	3.6
(4) Fuel Cycle Facility Safety	1.2	1.1	1.2	1.4
(5) Efficient Control and Chemical Systems	0.7	1.0	2.4	2.7
(6) Decommissioning	1.6	2.3	1.7	1.4
Total	27.8	34.2	38.2	40.8

4.6.1 MECHANICAL AND STRUCTURAL ENGINEERING

4.6.1.1 Summary and Comments

Load definitions and load combinations specification of effective earthquake motion; establishment of realistic damping; development of load combination methodology; specification of pipe break loads.

Structure/component Safety Margins Assessment of Soil structure interaction uncertainty; determination of steel containment failure modes and loading capacity; validation of computer codes; analysis and experiment of shear wall panel under seismic type loads; design criteria for auxiliary feedwater systems not seismically qualified; standard development on containment leakage testing criteria for anchoring supports to concrete and pile foundations.

NRC's program objectives are sound. Actual application to licensing and standards depends heavily on quality of results. Effective earthquake ground motions may have to be specified, case by case, because of uncertainties involved in soil structure interaction, nonlinear modeling and analysis. Damping recommendation requires careful investigation to ascertain the specific facility and configuration characteristics embedded in data. Sensitivity of seismic risk uncertainties may be dependent on reactor configurations. Equipment/component qualification Development of qualification and inspection methods, criteria, and standards for valves snubbers, pumps and other mechanical equipment and components; continued monitoring of the industry sponsored test program on safety and relief valves.

Increased involvement in foreign research dynamic characteristics of structures, equipment and components; seismic responses review; test data application.

4.6.1.2 Relationship to EPRI projects

EPRI's structure Integrity Subprogram projects have similar objectives as NRC's program. In many respects, both programs complement each other by having different test matrices and emphasis.

(1) RP964: in situ piping system damping and laboratory dynamic margin

(2) RP1324, 1733: pipe whip and jet loads

(3) RP810: soil structure interaction

(4) RP307,1550: nonlinear analysis methods development

(5) RP1554, 1707: TMI-2 Mechanical component recovery and extension of equipment qualification data bank to include mechanical components.

(6) RP1444: Foreign experimental data and analysis

4.6.1.3 Areas for possible EPRI and NRC cooperation

Close communciaton is needed to avoid undesirable duplications.

4.6.2 PRIMARY SYSTEM INTEGRITY

4.6.2.1 Summary and Comments

The scope of the USNRC Primary System Integrity Program parallels that of the EPRI Nuclear Systems and Materials System Integrity Program, although the USNRC budget is somewhat larger. The USNRC Program does include the Steam Generator Examination Facility and associated activities, which fall under the PWR Steam Generator Owners Group Program at EPRI; however, the EPRI System Integrity Program includes areas such as ASME Codes and Standards that are addressed in the USNRC Mechanical Structural Engineering Program. On balance, therefore, the EPRI budget is smaller by about 30%.

Major concerns addressed by the USNRC Program are pressurized thermal shock (i.e., overcooling transients with repressurization), safety margins on environmentally degraded or cracked piping, fracture toughness requirements for embrittled vessel welds and base metal, assessment of pipe cracking mitigation procedures, evaluation of industry ultrasonic inspection capability for both piping and near surface flaws in vessels, real time flaw detection, and acoustic emission monitoring of fatigue damage.

Near-term emphasis is being placed on the overcooling transient issue and on ultrasonic inspection of flawed piping. The latter will primarily be a review of industry round robin NDE results on both austenitic and ferritic piping, and the USNRC evaluation will determine in large measure, their posture toward the BWROG IGS'CC problem resolution and toward PWR feedwater nozzle cracking remedies.

4.6.2.2 Relationship to EPRI Programs and Projects

Of the major concerns addressed by the USNRC Primary System Integrity Program, almost all are directly or indirectly addressed in current or planned EPRI System Integrity Program research. The pressurized thermal shock test planned by USNRC for FY82 will be a useful case for application of the ductile fracture mehcanics methodology of RP1237 and some of the information from RP 1021 can be combined in order to address the concern about overcooling transients with repressurization in irradiation embrittled reactor vessel weldments.

The large-scale degraded and cracked pipe tests subject to thermal shock, waterhammer, and seismic loads planned for FY83 and FY84 will be factored into the planning for RP1543 Reliability of Piping and Fittings and some of the information may be relevant to tests conducted on large austenitic stainless steel pipes with IGSCC on the BWROG Program. The validation of fracture toughness requirements for irradiated, low shelf weld and plate is related to work being done on RP886-2 Evaluation and Prediction of Neutron Embrittlement in Reactor Pressure Vessel Materials, RP1021-4 (Fracture Toughness Correlations and Reference Curves), RP1240-1 (Collection and Evaluation of Data for Irradiated Pressure Vessels), and RP1326-1 (Development of a Crack Arrest Roughness Data Bank for Irradiated RPV Materials). The USNRC plans laboratory scale piping tests to evaluate measures proposed by vendors to reduce pipe cracking. These tests are related to similar evaluations being conducted within the EPRI BWROG Program.

Most of the NDE elements of the USNRC Program are related to projects in the EPRI Nondestructive Examination Subprogram. The round robin tests on improved ultrasonic flaw detection in stainless steel piping is related to ongoing work in the EPRI BWROG Program. Similar applications to carbon steel piping are not part of the EPRI System Integrity Program. The USNRC Program will be concerned with evaluating the level of improvement that can be obtained with advanced NDE techniques that may be proposed for incorporation into the ASME Boiler and Pressure Vessel Code, Section XI, 1964. Similar efforts are implicitly contained in RP1570, the NDE Center project description. The USNRC Program contains two large development efforts one on realtime SAFT UT and the other on acoustic emission that involve laboratory work, adaptation for field application, and field evaluation.

During FY82, in fact, a vessel fatigue and burst test is to be conducted using acoustic emission monitoring of prefailure crack ground. EPRI's System Integrity Program is related through RP 0608 (Acoustic Emission) and RP0606 (Acoustic Imaging). In addition, the EPRI Program will begin new work in Ultrasonic Near Surface Detectability that relates directly to the USNRC Program element on round robin test results for improved detection and recharacterization of near-surface pressure vessel flaws.

4.6.2.3 Areas for Possible EPRI and USNRC Cooperation

The EPRI System Integrity Program has been well coordinated with the USNRC Primary System Integrity Program for a number of years. The mode of cooperation has been through: (a) ASME Code activities, especially Section XI; (b) EPRI liaison on Unresolved Safety Issue (USI) Working Groups, such as that for USI A 11 (Reactor Vessel Material Toughness); (c) periodic EPRI technical information meetings, to which cognizant USNRC personnel are invited; and (d) personal contacts. Areas that could be strengthened include the BWROG Program and NDE acceptance criteria. More emphasis should be placed on interfaces and dialogue with personnel from the USNRC office of Nuclear Reactor Regulation, since critical decisions with respect to industry capability in NDE may affect the USNRC's willingness to accept concensus standards.

Major project areas for cooperation include: (i) RP1236-1 (Weld Repair of Heavy Section Steel Nozzles); (ii) RP1542 (Structural Reliability Methodology; (iii) RP1543 (Reliability of Piping and Fittings); (iv) RP2055 (Support and Structure Reliability); and (v) RP2056 (TMI-2 Pressure Boundary Requalification). It is our intent to maintain working relationships with USNRC personnel both in the office of Nuclear Regulatory Research and in the office of Nuclear Reactor Regulation in all of these project areas.

4.6.3 ELECTRICAL EQUIPMENT QUALIFICATION

4.6.3.1 Summary and Comments

The extensive NRC program renders the industry vulnerable to ratcheting An example is the finding of Sandia (based on limited test data) that the simulated age degradation of certain materials is highly dependent on the sequence of age conditioning test (e.g., thermal and radiation). This led to the blanket regulations that synergisms be addressed in all qualification programs. Fire protection testing to validate existing standards and provide basis for issuing a regulatory guide; qualification testing research to develop test methods and procedures for qualifying electrical equipment (includes evaluation of equipment from TMI-2, improved accelerated aging techniques, and a new high performance LOCA/HELB simulator at Sandia); classification of electrical equipment to clarify the scope of equipment covered by various NRC regulations.

4.6.3.2 Relationship to EPRI Projects

Equipment Qualification Data Bank and efforts to reduce the cost to utilities of mounting regulations (RP1707-3, Radiation Effects; RP1707-4 Aging/Seismic Tests; RP1707-5 Plant Environment Study). Research to advance technology of aging and qualification falls by default to NRC to support regulations.

The recently expanded EPRI TMI-2 Information and Examination Program includes projects aimed at using data from TMI to improve qualification methodology.

4.6.3.3 Areas for Possible EPRI and NRC Cooperation

Current cooperation consists of occasional EPRI NRC information exchange meetings and informal peer reviews. Areas for possible cooperation are those high-priority subjects that would benefit utilities, but are not being addressed due to the limited EPRI budget.

4.6.3.4 Specific Recommendations and their Justifications

Two areas appropratie for cooperative efforts are:

(1) Surveillance Techniques for On-Going Qualification: the development of in situ surveillance techniques geared to known aging degradation mechanisms could elimiate the need to qualify by test the large amount of accessible equipment in benign plant areas and guard against unexpected common mode failures.

(2) Quantification of Qualification Cost/Benefit based on Overall Risk Assessment: Many industry observers conclude that growing regulations and costs of equipment qualification cannot be justified in terms of their contribution to increased plant safety. A risk/benefit study could also prioritize equipment types so that limited qualification resources could be focused on high-payoff areas.

4.6.4 EFFLUENT CONTROL AND CHEMICAL SYSTEMS

4.6.4.1 Summary and Comments

Determination of source terms; measurements and evaluation of in plant measurements of effluent control systems; experiments concerning fission product and removal effectiveness; evaluation of on site radioactive waste handling systems, of hydrogen control and mitigation systems, and water chemistry characteristics of LWRs.

NRC's program is similar to EPRI's Subprogram Effluent Monitoring and Control, and Chemical and Radiation Control. EPRI's project does not address measurements of effluent control systems, but of effluents themselves. EPRI is not involved in readiolytically generated hydrogen, but is in that generated from degraded cores. Control and mitigation from the radiolysis process is well understood and little work needs to be done.

4.6.4.2 Relationship to EPRI Projects

(1) SOURCE TERM RELATED: RP1548 Study of fission product behavior in LW systems following abnormal occurences.
RP1558 Reactor system fission product pathways

(2) EFFLUENTS MEASUREMENT: RP2129 Isotopic Analysis of LWR stack effluents using Ca Te detectors

(3) FISSION PRODUCT REMOVAL: RP1548 "Study of Fission Product Behavior in LWR Systems Following Abnormal Occurences."

(4) ON SITE RADIOACTIVE WASTE HANDLING: RP1560 Investigation of Post Accident 85 Kr Effluent Control

4.6.4.3 Areas for possibel EPRI and NRC Cooperation

There has been cooperation in the area of source terms and radioactive materials. With EPRI's RP1558-1, "Reactor System Fission Product Pathways," it would be beneficial that NRC do leakage path ways and EPRI do system pathways.

Improvement of charcoal absorbers is an area where cooperation could be helpful.

4.6.5 DECOMMISSIONING

4.6.5.1 Summary and Comments

Expenditure of $5.5 million over the next 3 years to finalize criteria for determining whether sites are clean enought for decommissioning (termination of control by licensing) appears appropriate. Sound criteria will help industry, as well as NRC, and considerable technical support will be required to arrive at criteria which are justified, understandable, enforceable, etc.

There is some concern with the focus on release for unrestricted use. Industry's postion has been that NRC should give equal attention to the more likely alternative of nculear sites being utilized by owners for functions not requiring NRC licensing but not necessarily permitting unrestricted public access either. For example, conversion of a nuclear power plant to a fossil unit should not require to completeness of decontamination that subsequent siting of a supermarket or a hospital might require. Need for such alternative release conditions should at least be acknowledged by NRC, even if given lower priority.

4.6.5.2 Relationship to EPRI Projects

Study has been given in S&M Department to the "eternal plant" concept under which provision would be made at the time of design for facilitating utilization of site structures and support system after termination of initially licensed functions. Preliminary conclusions are that such an approach could be practical and cost effective. However, it would require that regulations permit something other than complete razing of licensed site facilties upon decommissioning, as implied by the current emphasis on unrestricted release

4.6.5.3 Areas for Possible EPRI and NRC Cooperation

There are no current EPRI projects directly related to plant decommissioning which could be integrated with NRC's program. However, they have included studies of any reactor or fuel cycle facility decommissioning work that is done. Where such work is undertaken by commercial power reactor owner/operators, it is likely that EPRI will be involved in providing technical support and/or evaluating data and experience gained at such time, planned integration of EPRI and NRC effort would be to the benefit of both parties. Also, if and when further work on longer term nuclear site utilization is undertaken, it should be coordinated with NRC consideration of appropriate clean up criteria, etc.

4.6.5.4 Specific Recommendations and their Justifications

NRC criteria development efforts should be followed to be sure that utility interest are at least considered and that needs for technical support of site disposition decisions and actions are identified well in advance.

4.7 WASTE MANAGEMENT

The waste Management decision unit includes three planned accomplishments. The budget for the three planned accomplishments are shown below:

Planned Accomplishments	1981	1982	1983	1984
(1) High Level Waste	5.4	13.5	9.6	9.6
(2) Low Level Waste	3.2	5.5	5.5	5.5
(3) Uranium Uncovery	1.8	3.1	4.5	4.5
Total	10.4	22.1	19.6	19.6

General Comments

(a) Relation of ONRR to ONMS&S - The material provided appears to represent programs for the Office of Nuclear Regulatory Research at the level of "decision unit justification," i.e., at the level of branches under the four main technical divisions. While the Office of Nuclear Regulatory Research has recently been reorganized to include the former Office of Standards Development, the technical assistance contracts in the Office of Nuclear Material Safeguards and Security (ONMS&S) contain the major activities leading to nuclear waste disposal and uranium mill tailings criteria, which are of major interest to the utility industry.

(b) Programmatic budget - contact with NRC staff indicates the coordinated program logic for waste management is described in a "programmatic budget" that defines the overall NRC activities in waste disposal now divided as follows:

(1) Office of Nuclear Material Safeguards & Security - "technical assistance" contracts, i.e., these support 10CFR60, 61

(2) Office of Nuclear Regulatory Research - research contracts

(c) Uranium mill tailings regulatory controversy - recently Congressmen (see Congressional Record, July 24, 1981) have introduced amendments to the 1982 NRC budget prohibiting expenditure of any NRC funds to enforce the mine and mill tailings regulations (45CFR65521 30, October 30, 1980).

Simultaneously there is a lawsuit now pursued by the American Mining Congress based on a suit initiated by Kerr McGee/Homestake/Anaconda that seeks to overturn the regulation, at least in part, on procedural grounds, i.e., allegedly NRC cannot issue their regulation until EPA has issued the appropriate regulation and EPA has not yet acted.

The present controversy in mill tailings is exemplary of what may result in LLW and HLW (10CFR60 and 61) if the regulations are not practical, cost effective, and related in a sound technical manner to real health and environmental effects.

The current issue is the effect of a 2 pico Curie/m^2 radon emmanation and the requirement to bury tailings with up to 3 meters of earth to reduce potential radon lung doses. There is a possibility that the current controversy will spill over into the comments on 10CFR61 and even 10CFR60, potentially delaying the waste disposal program.

(d) Recommendations - further coordination of EPA/NRC radiation regulatory efforts is required for a joint CPRI/NRC/EPA program in this aea to be effective.

4.7.1 HIGH-LEVEL WASTE

4.7.1.1 Summary

$9.6 million RY1983. Continue research and standards program for assessment of HLW geologic repositories.

4.7.1.2 Relationship to EPRI Projects

EPRI is conducting directly related and parallel effort under the following contracts:

RP1579 2	(Nuclear Safety Assoc.) Analysis/Criteria
RP1579 4	(The Analytic Science Corp.) Analysis/Criteria
RP1579 6	(Catholic University) Waste Forms Testing

4.7.1.3 Areas for Possible EPRI and NRC Cooperation

The total area might benefit from coordinated EPRI/NRC/EPA programmatic activity.

4.7.1.4 Specific Recommendations

None at this time. Current comments by industry on 10CFR60 will illuminate possible areas for NRC/EPRI cooperation.

4.7.2 LOW-LEVEL WASTE

4.7.2.1 Summary

$5.5 million FY1983. Continue research and standards programs for assessment of LLW disposal facilities.

4.7.2.2 Relationship to EPRI Projects

RP1579-5	(Rogers Assoc. Eng. Co.) Criteria and Analysis
RP2063-1	(Rogers Assoc. Eng. Co.) Criteria and Site Election
RP2063-2	(TASC) Criteria and Analysis
RP2130E	(to be selected) Characteristics of Plant Waste
RP2130E	(to be selected) Characteristics of Current LLW Forms

4.7.2.3 Areas for Possible EPRI and NRC Cooperation

The total area might profit from a coordinated EPRI/NRC review in cooperation with DOE's program through the Technical Management Center at EG&G, Idaho.

4.7.2.4 Specific Recommendations and Their Justifications

(1) Specific areas may be evident based on industry comments on 10CFR61 proposed regulations with comments due October 24, 1981.

(2) No specific areas for waste form and packaging cooperative research have been identified pending results expected fall 1981 from RP1579-5 and RP2063-1, -2.

4.7.3 URANIUM RECOVERY

4.7.3.1 Summary

$4.5 million FY1983. Continue and expand testing and assessment methods for characterizing geologic/hydrologic parameters for uranium facilities and tailings disposal methods.

4.7.3.2 Relationship to EPRI Projects

While there is potential to apply current RP1579 and RP2063 project results to uranium mill tailings, the only EPRI work on mill tailings was performed under RP767 3, completed in fall 1979, and summarized in NP-1128.

4.7.3.3 Areas for possible EPRI and NRC Cooperation

Work in this area will be confounded by the American Mining Congress lawsuit, current Congressional dissatisfaction with mill tailings regulations, and limitations on the EPRI budget in this area.

4.7.3.4 Specific Recommendations and their Justifications

Participation by EPRI would require policy decision by EPRI management, revision to one- and five-year plans, and additional staff and budget support.

4.8 FACILITY OPERATIONS AND SAFEGUARD

The Facility Operations and Safeguards decision unit consists of five planned accomplishments. The budget for the individual planned accomplishments are outlined as below:

Budget 1981-1984 ($M)

Planned Accomplishment	1981	1982	1983	1984
(1) Plant Instrumentation and Control	1.3	2.9	4.7	5.7
(2) Safeguards	5.5	6.6	2.7	2.7
(3) Human Engineering and Quality Assurance	3.9	3.3	6.3	7.9
(4) Occupational Radiation Protection	2.1	2.4	2.6	2.6
(5) Emergency Preparedness	0.5	1.2	0.5	0.2
Total	13.3	16.4	16.8	19.1

4.8.1 PLANT INSTRUMENTATION AND CONTROL

4.8.1.1 Summary and Comments

(1) Instrumentation and Control Program From the one line descriptions of accomplishments, it is difficult to reconstruct a top level coordination plan. Such a plan could be based upon an operational plant model that identifies critical plant safety and availability requirements and then relates plant subsystems to these requirements. This model would provide a technical framework for identifying:

(a) Systems requiring more or better instrumentation,

(b) Implication of control systems on safety and availability,

(c) Performance, testing and maintenance requirements,

(d) Application needs for noise analysis in surveillance application,

(e) Self-check and redundancy requirements.

The use of dynamic simulation and reliability assessment tools on a comprehensive basis across the plant would provide a strong technical foundation for the coordination plan. The 1984 accomplishment pertaining to hybrid simulation of representative control systems does not appear to be sufficiently comprehensive. In summary, the plan proposes many valuable bottom level accomplishments, but lacks an overall integrated plan which might help to put specific efforts in the proper perspective. The industry needs to focus more attention on a systems approach aimed at increasing plant safety and availability. The NRC research program should help to promote a more integrated approach.

(2) Human Engineering

The projects within this program appear to be better integrated; however, there does not appear to be much coordination with the Instrumentation and Control program. I think that the divisions of responsibility between control systems and operators, plus the impact of one on the other deserves more emphasis.

4.8.1.2 Relationship to EPRI Projects

The RP891 project includes the following that are related to the NRC plan:

(1) Graphic display capability will exist by September 1981,

(2) SNUPPS evaluation for measuring human performance capability,

(3) Study the effects of factors affecting human performance for four plants,

(4) Characterization of symptom-oriented procedures within DASS

4.8.1.3 Areas for Possible EPRI and NRC Cooperation

Although there is considerable technical overlap between EPRI and NRC projects, in most cases we are approaching the problem from different perspectives. Generally speaking, this restricts the potential for collaboration, however, the Phase III DASS project might be a suitable candidate for technical cooperation for the following reasons:

1. NRC R&D appears committed to an assessment of the Disturbance Analysis and Surveillance System (DASS) (FY84). EPRI should collaborate in ensuring adequate review and assessment.

2. The DASS effort provides an integrated technical framework for coordinating I&C and computerized operator aids developments and evaluations. NRC involvement would assist future development.

4.8.2 SAFEGUARDS

4.8.2.1 Comments

(1) The 1973 NRC budget in safeguards is based on a presumption of no reprocessing. In light of recent Reagan administration changes, there is reason to reconsider the basis for this planned level of effort.

(2) The material accounting and physical protection requirements for LWRs now require a work force approximately equal to the plant maintenance and operations force, i.e., 180 to 250 employees. It would be appropriate to recommend re-evaluation and restructuring of the NRC program to develop more cost effective means of meeting appropriate safeguards and physical protection requirements.

4.8.2.2 Relationship to ERPI Program

EPRI conducts no safeguards or physical security research. Such factors are evaluated in conjunction with work on two other areas, (1) diversion resistant fuel cycle facilities (RP1578) and (2) plant physical protection in the improved LWR effort (RP1585).

4.8.3 HUMAN ENGINEERING AND QUALITY ASSURANCE

4.8.3.1 Summary and Comments

The terseness of stated plans does not give a good picture of the priority of each objective nor the level of effort devoted to it. This incompleteness points up the need for periodic interface meetings (e.g., semiannually) of the type held on July 2, 1981 between the NRC, DOE, INPO and EPRI. Such meetings afford the opportunity for more detailed information so as to assure that on-going R&D activites of the respective organizations are technically feasible, needed, and are complementary rather than duplicative in nature.

The rationale for the nature of the projected NRC program is sound and is well substantiated by prior events and investigations. The concept is sound that a technical basis must be further developed to adequately support current and anticipated regulatory needs. Regulations issued without the qualifying R&D might be unnecessarily burdensome on the industry without commensurate enhancement of system safety and effectiveness.

The sequential process of R&D followed by regulations is well illustrated by the proposed approach in selection and training, computerized display systems, uses of training simulations, formulation of human engineering design standards for reactor control rooms, and systematic experiments on working hours and stress levels, and the effectiveness of the operator licensing process. All of these efforts are legitimate concerns and worthy of systematic R&D. A good example of an NRC objective which lacks sufficient information for an adequate response is the following 1984 objective: "complete the planning and begin to implement a long-term program of monitoring the effectiveness of operating personnel."

4.8.3.2 Relationship to EPRI Programs and Projects

There are several EPRI projects that have been completed, (C), are on-going, (O), or planned, (P) that represent potential inputs of a complementary nature to NRC objectives:

(O) RP769, Performance Measurement System for Training Simulators Identify techniques of quantitatively measuring human performance capabilities

(O) RP891, DASS Develop regulatory guidance and evaluation criteria for computerized display systems and computer-based diagnostic systems

(P) RP2011, Development of an Alarm System Improvement Guide Complete a graphic/display research facility -- on human performance as effected by information display content and format.

(O) RP1637, Human Engineering Guidelines (C) RP1126, Human Factors Review of Power Plant Maintainability

(C) RP501-3, Human Factors Methods for Nuclear Control Room Design

(C) EPRI Special Report NP-1918, Anthropometric Data Base for Power Plant Design. Develop regulatory guidance for integrating human factors engineering methodology into system design.

(O) RP1396, Test of Job Performance Aids (JPAs) for Power Plants Develop regulatory guidance on the content and format of nuclear power plant operating procedures.

(P) RP2009, Nuclear Power Plant Communications Stands Complete assessment of design standards for reactor control rooms and incorporate into regulatory guidance.

(O) RP769, Performance Measurement System for Training Simulators Develop the technical basis for regulatory guidance to improve the capabilities and uses of training simulators.

(O) RP1637, Human Engineering Guidelines Complete a critical review of the basic human factors engineering data underlying existing human factors standards and recommend necessary improvements.

4.8.3.3 Areas for Possible EPRI and NRC Cooperation

In addition to the potential linkage of projects listed immediately above, other areas for possible EPRI and NRC cooperation include: Maintainability Studies

Work Structure; e.g., working hours, stress levels, etc., Quality assurance requirements

Develop task analyses for operations and support personnel

Qualification requirements of instrument and control technicians Selection, education and training requirements of operations and support personnel

4.8.3.4 Specific Recommendations and Justifications for the Recommendations

(1) Many of the NRC objectives are characterized by interrelated tasks and requirements. It would be advantageous, on a selected basis, to allocate responsibility for the component parts to NRC and EPRI with shared results. NRC would have final responsibility for the determination of whether the R&D results should be translated into requirements. A mechanism for allocation of task responsibility could be the interface meetings mentioned earlier.

(2) A systematic R&D report exchange program should be established. At the present time, this exchange program is rather informal in nature.

4.8.4 OCCUPATIONAL RADIATION PROTECTION

This area is very difficult to assess - not enough detail provided on content. Air sampling is an area worthy of more work. Not sure neutron dosimetery is a real problem.

4.8.5 EMERGENCY PREPAREDNESS

4.8.5.1 Summary and Comments

Based on the statements of planned accomplishments, the program plan has the following objectives:

1983 -- Develop proposed revisions to the regulations governing power plant licensee emergency planning.

-- Develop a suite of accident scenarios for use in testing emergency plans, primarily offsite plans.

-- Develop Regulatory Guides for emergency planning by fuel cycle facility licensees.

1984 -- Develop effective (final) revisions to the regulations governing power plant licensee emergency planning.

-- Develop draft Regulatory Guides for using the accident scenarios.

Further, the remarks justifying the program mention incorporating "implementation and licensing experience on the adequacy of nuclear facility emergency preparedness" and examining "interfaces among NRC and other federal agencies and state governments during emergencies...to assure responsibilities are clearly defined and that communications and facilities support the role of each agency."

The program budget is expected to decline from $1200K in FY82 to $500K in FY83 to $300K in FY84. The decline is appropriate, but the magnitudes may still be high. Development of the suite of accident scenarios began, under contract, in 1979 or 1980 and it is expected to drag on into 1983. The current emergency planning regulations went into effect in 1981, so the revisions are not expected to be extensive in 1983.

Finally, evaluating and incorporating implementation and licensing experience is not a research function; it should be the responsibility of the Emergency Preparedness Program Office (EPPO) within IE. Similarly, interfaces between NRC and other federal or state agencies are primarily the responsibility of the Federal Emergency Management Agency more than NRC. Again, the NRC involvement should be through EPPO, not RES.

4.8.5.2 Relationship with EPRI Projects

In regard to EPRI/NSAC programs and projects in this area, some preliminary work has been doen by SRI ("A Preliminary Examination of Emergency and Crisis Decision-making," December 1979), and proposals received from SRI for extending this effort. At present, negotiation with potential contractors on appropriate projects in emergency management, including extension of the decision-making work are underway. These areas of interest all relate in some measure to the degraded core work underway in NSAC and include the following:

-- Studies on possible disconnects between remedial actions taken to prevent or mitigate degredation of nuclear power plant systems and emergency actions taken to identify off-site risks and to prevent or mitigate off-site impacts to health.

- Evaluation of present emergency management chain-of-command structures in and among reactor operations and state and federal organizations for decision-making practicality and operability, and determination of the appropriate information requirements of these various entities from the standpoint of reactor operator responsibility.

- Studies on information links and other communication requirements, from the standpoint of need for decision-making on the part of nuclear power operations.

- Determination of appropriate levels of initiating emergency management response on the part of nuclear power operations in regard to potential and real levels of health risk, on-site and off-site.

- Review of the bases for classifying nuclear power operation emergencies, including both risks to plant and risks to health.

- Review of guidance to licensees on emergency management actions for the protection of the health and safety of workers and of the general public, for appropriateness, completeness, practicability and utility; development of decision-making guidance for plant operations for evaluating and recommending appropriate health protective actions.

- Studies on the relationship between degraded core and other accident assumptions and emergency preparedness requirements from a sensitivity analysis basis.

- Need for and conduct of appropriate seminars on the technical bases for nuclear crisis management.

- Studies to identify possible restrictions placed by state or federal authorities on the basis of control of off-site emergency conditions that could preclude actions necessary to prevent or to mitigate adverse reactor system degradation.

4.8.5.3 Areas for Possible EPRI and NRC Cooperation

In regard to areas of possible EPRI/NSAC cooperation with NRC, while there appear to be none at the moment, we should be developing a much closer rapport with NRC, FEMA, the National Governor's Association, INPO, AIF and EEI on these matters. The present approach of the NRC to emergency management is "heavy," to say the least, with many manifestations of overkill or near-overkill. In all honesty, there may be a huge curriculum of emergency response presently established, but it appears that the net result in event of a future emergency may be no better than TMI-2. EPRI may be in the unique position to offer a fairly unbiased critical approach to assessing what is being done and what is actually needed.

4.8.5.4 Specific Recommendations and Their Justifications

(1) RES should more closely coordinate its activities with (perhaps take more direction from) EPPO.

(2) The only facet of the degraded core rulemaking pertinent to the emergency planning rule and the accident scenarios will be the improved assessments of the fission product source term: the nature, magnitude, and probability of fission product releases from containment. These improved assessments should be reflected in the accident scenarios and the requirements of the emergency planning rule.

(3) The justification given by NRC for its emergency preparedness R&D program seems to be focused wholly on the needs of the regulators and on enlarging the regulatory process. The major deficiency is that the needs of the licensees and the licensee/regulator interfaces are missing as components of this R&D program. More attention is needed on the technical bases for emergency preparedness requirements, on the needs of the licensed activities and on the interfaces between licensees and federal and state agencies. NRC also needs to make much clearer the distinction between preaccident emergency preparedness requirements and those requirements that would be operative for the actual management of a real emergency. It appears that NRC runs a very real danger of developing a "cookbook" of constraints over licensees and state and local authorities through the issuance of the preaccident NRC/FEMA/licensee/state and local authorities emergency plan that the actual authorities, i.e., licensee management and state Governors, during the real emergency may come into conflict, with disastrous results for public health and safety and confidence. Some portion of the NRC R&D budget should be devoted to these issues, including:

-- Development of guidance and technical and procedural support for licensees,

-- Studies on emergency standards,

-- Studies on interfaces and communications between licensees and regulators during emergencies, and

-- Studies on institutional and operational factors essential to good emergency management decision-making among licensees and state and federal authorities.

(4) It is not clear why it is appropriate that the program budget for emergency preparedness decline from $1200K in FY82 to $500K in FY83, and to $300K in FY84. Projected expenditures for accident evaluation and mitigation are expected to rise, conversely, from $35.6M in FY82 to $50.4M in FY83, and to $59.3M in FY84. The expected outcome of this latter effort would be better technical bases for emergency preparedness, and it would appear logical that that portion of the budget for emergency preparedness regulation track the research effort so as to fully utilize the knowledge gained.

5) One of the program plan objectives relates to the development of accident scenarios for exercising the emergency plans. The actual budget document text refers to "appropriate" accident scenarios; however, it is suggested that the word "realistic" would be a better modifier. As things now stand, nuclear power licensees have 1 or 2 major off-site emergency exercises each year. It is understood that NRC insists that all phases of emergency planning be exercised annually: this means that a variety of accidents and responses, including evacuation, must be combined in a rather unrealistic manner that is confusing to the emergency management staff at the site, and that is misleading if not demoralizing to the near population off-site. NRC should examine its posture in this regard, particularly from the standpoint of Commission policy from which direction on appropriate emergency drills can be drawn.

4.9 ADVANCED REACTORS

The Advanced Reactors decision unit comprises two planned accomplishments. The list of the planned accomplishments and their associated budget are indicated in the following:

Budget 1981-1984 ($M)

Planned Accomplishment	1981	1982	1983		1984	
			EDO	Set Aside	EDO	Set Aside
(1) Fast Reactors	10.3	0	0	21.0	0	24.0
(2) Gas-Cooled Reactors	2.8	2.0	2.5	0	2.5	0
Total	13.1	2.0	2.5	21.0	2.5	24.0

4.9.1 FAST REACTORS

4.9.1.1 Summary and Comments

In the area of fast reactors the current estimate is that there will be no effort in FY1982. But the FY'83 and '84 program is quite sizable ($21M and $24M) respectively. The program is geared to licensing issues and places major emphasis on investigations of core disruptive accidents, their phenomenology, analysis, consequences and mitigation. Large experimental programs are planned mostly at the Annular Core Research Reactor (ACRR) at Sandia Laboratories, the Large Melt Facility (LMF) and the FAST facility for sodium tests. Analytical work focuses on integration (Integrated Severe Accident Analysis code) and verification of existing codes for safety and accident analysis (SIMMER, SSC, and COMMIX) and for aerosol release and transport (QUICK and HAARM-3). A risk assessment program and reliability analysis is also envisioned for FY'83 and '84.

As the titles above signify, the research plan is geared towards evaluation of severe accidents; in particular the hypothetical core disruptive accident (HCDA). The only exceptions are two code development efforts, namely, the COMMIX and SSC codes. The former code is a general purpose code for single phase flow transients and the latter models LMFBR plant transients.

The accident energetics program includes experimentation with the annular pulsed reactor providing data for the verification of the SIMMER code. This code attempts to model the HCDA scenario within the vessel. In addition an integrated severe accident analysis code is in the plan for FY1984, wherein the progression of the core melt will be tracked up to containment challenge

The system integrity program is concerned with the interaction of core debris with vessel materials and containment material. Data has been obtained on the interaction of sodium and molten core materials with concrete and retention materials.

The ART program will be completed in FY1984. The end product will be the codes QUICK and HAARM-3, which would be benchmarked against some tests on aerosol transport.

The criticism one could voice for this program is that too much emphasis is placed on the hypothetical core disruptive accident with all its wide and almost unmanageable ramifications. Not enough attention is paid to less severe transients and on prevention. However, these latter objectives may be more appropriate for DOE and the nuclear industry.

4.9.1.2 Relationship to EPRI Projects

EPRI has contributed in the past few years in the development of the Prototype Large Breeder Reactor, a design oriented toward high safety and reliability standards. Code work has also been completed for LMFBR operational transients for both pool and loop designs. The current and immediately foreseeable EPRI program is very limited due to funding restrictions. The on-going projects focus on design reviews, operability and maintainability studies and risk comparison between LWR and LMFBR. A study for an integrated computer code for flow distributions in heterogeneous core LMFBR with orificed zones and dual plena is being contemplated. The assessment of existing safety and accident codes is also being considered.

There is practically no relationship to our work, since our LMFBR safety work have concentrated on preventive measures, e.g., heterogeneous cores, self-activated shutdown systems. We have not delved into the HCDA progression analysis and experimentation. The COMMIX code developed under NRC program has been employed in one of our projects for mixing studies in the lower plenum of PWR.

4.9.1.3 Areas for Possible EPRI and NRC Cooperation

It is possible to see that participation in some of the experimental programs might give EPRI a certain leverage in the planning process (similar precedents exist in LWR research area). These programs are, of course, large and a considerable contribution would be necessary for any significant EPRI participation. In the code validation and integration area experience has shown that EPRI and NRC perceptions and emphasis are quite different and the optimum results are obtained if they follow their separate paths. However, it is essential to maintain a level of mutual awareness of each other's efforts for possible constructive inputs.

The NRC sponsored work and the advanced reactors research plan may be useful in the LWR degraded core coolability and retention work recently initiated by EPRI and being organized presently by Technology for Energy Corporation (TEC) in the IDCOR program. This is perhaps the main area of cooperation. Other cooperation is possible and has been exercised before in the development of BODYFIT code at ANL, which is a successor of the COMMIX code.

4.9.1.4 Specific Recommendations and Their Justifications

(1) EPRI should be cognizant of the HCDA related work and its findings relative to licensing of LMFBR in the future. The relationship of the work pursued to the IDCOR work is interesting and may be of benefit. This should be explored.

(2) In order for EPRI to have a substantial input and perhaps influence in the dialogue with NRC it must maintain at least a minimum credible R&D program in this area. Despite lack of a firm commitment and schedule for the next large LMFBR plant in the U.S. a considerable effort is being planned by the federal government. A number of safety and engineering issues remain and their further elucidation will be very helpful in the design approaches, regulatory resolution and even public policies toward a perception of importance: of a very high plant reliability; of events of higher probability and lesser consequence; of the relative safety advantage of the LMFBR vis-a-vis other reactor technologies; and of a realistic understanding of the physical phenomena that are likely to occur in the plant. For these various risk assessment studies of both pool and loop, analytic development and assessment as well as new analytic capability must continue. Small studies to determine optimal configuration of important safety issues, such as the self actuated shutdown system may have a significant effect on the reliability of shut-down and hence eventually in the licensing process.

4.9.2 GAS COOLED REACTORS

4.9.2.1 Summary and Comments

The planned expenditures for FY'83 and '84 are about $2.5M up for $2M in FY'82. The program will concentrate on developing the necessary licensing data base and analysis methodology in anticipation of gas-cooled steam cycle/cogeneration plant application in FY1983. The program includes high temperature materials research, containment integrity and failure mechanisms, fission product behavior, and risk assessment. From a regulatory point of view this seems a reasonable and well balanced program.

4.9.2.2 Relationship to EPRI Projects

EPRI has supported a number of small projects for the examination of Peach Bottom-I fuel, the analysis of operational experience from Fort St. Vrain and the evaluation of technical and commercial feasibility. No projects are planned in the immediate future.

4.9.2.3 Areas for Possible EPRI and NRC Cooperation

An exchange of information from the two programs would be beneficial. The NRC could benefit from EPRI's emphasis on operational experience and data, whereas NRC supplies a substantial amount of materials test data.

4.9.2.4 Specific Recommendations and Their Justifications

If one assumes that a license application is likely by 1983 EPRI should maintain a cognizance of the prevelant licensing issues and be prepared to contribute the point of view of the plant operator at the appropriate time.

QUESTION 11. Should NRC be doing research to develop operator procedures for severe accident situations or should they be requiring industry to develop procedures to be validated by NRC?

ANSWER.

Nuclear Steam Supply System vendors and Owners Groups have or are developing guidelines for writing emergency operating procedures. The individual utility then uses these guidelines to write its plant-specific procedures. NRC research is not developing procedures but has developed a methodology to evaluate and validate emergency operating procedure guidelines. The first application of the methodology was to the Westinghouse Owners Group guidelines. Currently, the methodology is being applied to validate the General Electric BWR guidelines. In FY 1982, the methodology will be applied to validate the two remaining types of guidelines, the Combustion Engineering PWR and the Babcock & Wilcox PWR.

QUESTION 12. Despite the programmatic agreement on improved reactor safety between DOE and NRC signed on December 28, 1979, we found the two agencies not in full accord concerning their appropriate interface. Would you describe the interface specified in the "programmatic agreement"? How useful is this agreement in making consistent day-to-day decisions on research responsibilities at the two agencies?

ANSWER.

The interagency agreement on improved reactor safety called for the establishment of a high-level "Joint Coordination Group", for DOE to budget the improved reactor safety work, and for DOE to set up an appropriate field organization to implement the effort. Before the "Joint Coordination Group" could be established, DOE's light water reactor research budget request was reduced by about 2/3 allowing DOE to continue only essential ongoing programs. The projected improved reactor safety program has not yet been initiated. In December 1980, the Congress passed PL 96-567, the Nuclear Safety Research, Development, and Demonstration Act of 1980, which generally encompassed the improved reactor safety work.

Due to the limited funding received by DOE to implement the agreement, we are unable to say how useful it might have been in making day-to-day decisions; however, it appears to be an appropriate model on which to base future agreements with DOE.

QUESTION 13. Does this agreement cover advanced reactors as well? If not, what is the agreement between the agencies on their respective roles in safety research for advanced reactors? Where is this agreement documented?

ANSWER.

The programmatic agreement on improved reactor safety between DOE and NRC signed on December 28, 1979, does not cover advanced reactors. Informal contracts with DOE to develop a Procedural Agreement on Fast Breeder Reactor Safety Research were initiated last year and extensive discussions have been held. A draft agreement is currently under negotiation. A fact that must be recognized in the CRBR proceeding is the very formal relationship between NRC and DOE as an applicant. This defines DOE's responsibility to completely support their application.

As a result of points raised by you earlier in these hearings, DOE and NRC staff are proceeding to work out a similar agreement for the HTGR.

QUESTION 14. In the area of nuclear waste management, you state that "NRC work is controlled to avoid unnecessary duplication of DOE work." Yet presumably the scope of NRC work is not simply defined according to where DOE stops working. What is the agreement between NRC and DOE on their respective roles for waste management research? Where is it documented?

ANSWER.

The research programs of DOE and NRC reflect their statutory role: for DOE, of designing, constructing and operating a high-level waste repository and, for NRC, of licensing and regulating to protect the health and safety of the public. DOE research generally is developmental in character and is done to evaluate candidate materials and designs in support of their responsibility for design and construction of a repository. On the other hand, NRC research is aimed primarily at providing NRC with an independent capability for assessing the completeness and validity of the DOE predictions of long-term waste isolation and safety performance.

There is in existence a formal agreement (copy enclosed) between the two agencies for the full exchange of program information including research. Pursuant to this agreement, NRC provides to DOE program descriptions (Statements of Work) for all of its high-level waste contracts and NRC staff visit DOE contractor facilities to exchange technical information. The DOE/NRC agreement explicitly provides that the exchange of information is done with the intent of avoiding unnecessary duplication of research and development activity.

2/6/81
406.1.2

DOE-NRC PROCEDURAL AGREEMENT
FOR THE COORDINATION OF DOE AND NRC
HIGH-LEVEL WASTE PROGRAMS

1. Purpose

The purpose of this Agreement is to formalize a relationship between the Department of Energy (hereinafter called the Department) and the Nuclear Regulatory Commission (hereinafter called the Commission) that permits the Agencies to coordinate their commercial high-level waste programs without compromising the Commission's ability to independently license and regulate future commercial high-level waste storage and/or disposal facilities of the Department.

Implementation of the actions identified below should be done with the intent of (1) having a complete and full exchange of information and data on each Agency's commercial high-level waste management program, (2) avoiding unnecessary duplication of commercial high-level waste management research and development activities, (3) reducing expenditures for waste management programs by clarifying each Agency's programs, and (4) being responsive to the other Agency's needs through future program planning.

2. Actions

The following actions will formalize the relationship between the Department and the Commission with respect to their respective commercial waste management programs. While this procedural agreement establishes a formal relationship between the two Agencies, it is not intended to inhibit the flow of technical information and exchanges between the Agencies.

a. Meetings

The Department and the Commission shall conduct periodic meetings on specific topics important to the two Agencies in order to enhance the exchange of information and avoid duplication of effort. These topics include siting, repository design, waste package development, performance assessment, environmental impact assessment, and schedules for licensing activities. At the beginning of each fiscal year, the Interface Coordinators (see Section e.) will jointly prepare a proposed schedule for formal meetings for the coming year. These meetings will include status reports on major activities being pursued at each of the Agencies in these areas. The staff of the Commission shall conduct periodic meetings on its technical program and the status of its regulations, regulatory guides, and licensing policies. The staff of the Department will conduct periodic meetings on their technical program and the status of the reports and applications which they anticipate submitting to the Commission. Both Agencies will also present schedules for completion of major milestones for their various programs and an overview of long term plans. The date and agenda for formal meetings conducted by the Commission for the Department shall be mutually agreed upon by each Agency. The Commission will make requests of the Department for formal meetings at least six weeks in advance and an agreed upon agenda will be established at least four weeks in advance of each formal meeting. Exceptions to these scheduling requirements will be considered on a case-by-case basis.

The meetings will be timed to properly fit the Department and Commission budgetary cycles such that future courses of action can be changed readily if deemed necessary to avoid unnecessary duplication of effort and expenditure of funds. The public will be permitted to attend these meetings as observers if they so desire.

In addition to the above meetings, there shall be periodic informal meetings between the Commission staff and their contractors and the Department staff and their contractors to facilitate exchanges of current technical information. Written requests for meetings will be submitted to the Interface Coordinator (see Section e.) at least four weeks in advance and will include a sufficiently detailed description of the topics to enable preparation of an agenda. An agreed upon agenda (and meeting date) will be established at least two weeks prior to each meeting. Exceptions to these scheduling requirements will be considered on a case-by-case basis.

Requests for meetings between the Commission staff and the Department staff shall be in writing to the Interface Coordinator and shall contain a sufficiently detailed description of the meeting topics to enable preparation of an agenda. Contractors may participate in such meetings as observers.

At the completion of all meetings between staffs of the Department and the Commission, a meeting summary will be prepared by the meeting chairman or his designated representative and cosigned by the senior representative present from each Agency within two weeks of the meeting. This summary will be placed in the public document file and noted in the Federal Register. The Commission staff will be responsible for the preparation and filing of the Federal Register notice.

b. Exchange of Contracting Documents

The purpose of the exchange of contracting documents is to provide each Agency with a detailed understanding of all technical work that is being performed for the other Agency by its contractors. This exchange of information should minimize duplication of research activities and the resultant expenditure of funds.

A copy of all relevant Commission requests for proposals (RFP's) will be provided to the Department when submitted to the Commerce Business Daily. When contracts are awarded, a copy of the scope of the contract, along with the identification of the contractor and principal investigator, will be forwarded to the Department. The Department will provide the Commission with a copy of all relevant preacquisition plans for each contracting action prior to announcing the availability of the RFP in the Commerce Business Daily. Upon written request by the Commission, the Department will provide the Commission with RFP's. When contracts are awarded by the Department, the Department will forward to the Commission a copy of the scope of work for the contract, identification of the contractor and the name of the principal investigator.

A copy of all Statements of Work which the Commission submits to the Department Operations Offices will be simultaneously provided to the Department Headquarters and the Department will provide the same information to the Commission.

c. Document Distribution

Document distribution lists of the Department and the Commission for nuclear waste isolation reports shall be developed into common distribution lists so that recipients, including Department and Commission contractors, will receive reports from both Agencies.

d. Staff Interactions

The Commission's staff and the Department's staff may participate as observers in technical oversight activities that are mutually agreed to by the Commission and the Department. Upon written request and subsequent agreement, each Agency will review and provide a timely written response to technical reports within the sphere of these technical oversight activities.

e. Interface Coordination

Each Agency will designate one cognizant individual called the Interface Coordinator who will be assigned responsibility for implementation of this agreement. This individual will be the principal contact between the Department and the Commission for his assigned activity and will be responsible for maintaining current and accurate records of Department-Commission actions carried out under this agreement.

f. Amendments

This agreement may be modified or amended in writing if agreed to by the Department and the Commission.

C A Heath 3/6/81

Colin A. Heath, Director
Office of Waste Isolation
Nuclear Waste Management Programs
Office of Nuclear Energy
U.S. Department of Energy

JBMartin 4/27/81

John B. Martin, Director
Division of Waste Management
U.S. Nuclear Regulatory Commission

QUESTION 15. Please supply us with copies of current memoranda of understanding between DOE and NRC on LWR safety research, CRBR research and waste management research.

ANSWER.

As discussed in Question 13 and 14 NRC is currently well along in working with DOE in developing agreements on fast breeder reactor safety and waste management. Question 12 discusses the status of the LWR improved reactor safety memorandum of understanding. Attached is the agreement between DOE and NRC on the LOFT facility and LWR Improved Reactor Safety.

December 28, 1979

DOE-NRC Interagency Programmatic Agreement
In Support of
Improved Reactor Safety

I. Introduction

The DOE-NRC Memorandum of Understanding executed February 24, 1978, establishes an overall management policy for the DOE and the NRC with regard to interagency relationships in the conduct of research programs and related activities. The policy statement ensures priority, independent access, and cooperation, and notes in Article 1 A that interagency agreements will be established for the major programs as defined by program offices in each agency constituting that interface. Article II B 1 notes that the Under Secretary, DOE, and the Chairman, NRC, will establish any additionally required policy for the conduct of interagency interface, relationships, and responsibilities.

II. Background and Specific Policy

On July 27, 1979, the DOE Under Secretary and the NRC Chairman agreed that DOE should undertake and conduct certain work directly responsive to the NRC Improved Reactor Safety Program. DOE will budget and fund for all agreed upon activities, in accordance with OMB guidance, and which are of significant programmatic benefit to DOE.

III. Purpose

The purpose of this agreement is to implement the policy noted in Article II, to establish working relationships, and effective lines of communication.

IV. Conduct of Work

A. General

1. A "Joint Coordination Group for Improved Reactor Safety" is established to review plans, expenditures, and progress of the work covered by this agreement. The review will include technical, programmatic, administrative and any other aspect which might affect the nature or progress of the work. The group will coordinate the research so as to optimize the utilization of all resources, and to initiate actions for appropriate improvement in the program.

   The joint coordinating group is comprised of the Deputy Assistant Secretary for Nuclear Reactor Programs, DOE; the Director, Office of Nuclear Regulatory Research, NRC; and other designated representatives of DOE and NRC. This group shall report progress at the DOE-NRC coordination meetings and at the DOE-NRC semiannual management meetings.

2. DOE will budget for and provide the authorizations necessary for the initiation and execution of specific agreed upon tasks subject to funding availability. NRC will assist DOE in budget preparation and justification of this activity to OMB and the Congress.

3. The work undertaken in support of this agreement will be included within the DOE Light Water Reactor (LWR) Safety Technology Program and will be administered by the Technology Management Center (TMC) for DOE.

4. DOE will authorize performance of agreed upon tasks through the appropriate DOE field office (contracting officer).

B. Specific

1. NRC will provide programmatic guidance, including desired end product(s) for each agreed upon task. These will be provided to the Deputy Assistant Secretary for Nuclear Reactor Programs, DOE, or his designee with information copy to the DOE contracting officer and to the TMC. This guidance will be evaluated by DOE (and the TMC) to assure that agreed upon work tasks are not unnecessarily duplicative of other ongoing DOE LWR Safety Technology Program activities, and to develop a definitive work scope, plan, and schedule for each agreed upon task.

2. DOE will provide definitive work scopes, plans, and schedules for NRC review and comment at least 15 working days prior to each authorization to proceed.

3. DOE-HQ will formally transmit agreed upon work scope, plans, and schedules and the necessary authorizations to the DOE contracting officer for implementation. Information copies will be provided to the NRC and the TMC.

4. Any guidance to the performing organization(s) which could change agreed upon costs, scope, or schedules will be provided only through the DOE contracting officer or his designated representative.

5. DOE, NRC, and TMC will assign specific personnel to this program. These representative will meet as required and discuss work performance, changes in scope, schedules, etc. to assure that the results are directly applicable to improved reactor safety.

VI. Reports and Reviews

A. Reporting Requirements - Progress and accomplishments will be included in the DOE LWR Safety Technology Program bimonthly reports, which will be distributed routinely to NRC.

B. Interim and Final Technical Reports - Results of each task will be reported in interim and final technical reports as specified by the individual statements of work in accordance with established DOE and NRC procedures.

C. Program reviews will be scheduled by the joint coordination group at times commensurate with progress of the work activity or as mutually agreed by NRC and DOE. These reviews will be documented and presented to both agency's principals.

VII. Limitations

Nothing herein is intended to conflict with the ongoing NRC Reactor Safety Research Program or the DOE Light Water Reactor Safety Technology Program.

VIII. Agreement

This agreement is effective upon acceptance and signature by the officials noted for FY 1980. Amendment to this agreement must be jointly approved by both agencies for future years or if funding responsibility changes.

FOR DOE:

Acting Deputy Assistant Secretary
for Nuclear Reactor Programs

12/28/79
Date

FOR NRC:

Director, Office of Nuclear
Regulatory Research

12/29/79
Date

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 2 9 1980

Mr. Robert L. Ferguson
Deputy Assistant Secretary for
Nuclear Reactor Programs
U.S. Department of Energy
Washington, DC 20545

Dear Mr. Ferguson:

Enclosed are the minutes of the April 16, 1980 meeting of the DOE-NRC Joint Coordination Group for Improved Reactor Safety. The meeting was most helpful in reaching a mutual understanding of each agency's position and efforts on research related to improving reactor safety. I look forward to continuing our cooperation in this area.

In accord with the proposal made at the meeting, I agree that the term of our existing DOE-NRC Interagency Programmatic Agreement in Support of Improved Reactor Safety be extended until September 30, 1981. If you agree with this proposal, please so indicate by signing below and returning one signed copy to me.

Sincerely,

Robert J Budnitz

Robert J. Budnitz, Director
Office of Nuclear Regulatory Research

Enclosure: Minutes

cc: R. Barber, DOE
M. Norin, DOE

Robert L. Ferguson
Deputy Assistant Secretary for
Nuclear Reactor Programs
U.S. Department of Energy

MEMORANDUM OF UNDERSTANDING
BETWEEN
THE ENERGY RESEARCH AND
DEVELOPMENT ADMINISTRATION
AND
THE NUCLEAR REGULATORY
COMMISSION

MEMORANDUM OF UNDERSTANDING BETWEEN THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION AND THE NUCLEAR REGULATORY COMMISSION

William A. Anders
Chairman

Robert C. Seamans, Jr.
Administrator

Aug 8, 1975
Dated

MEMORANDUM OF UNDERSTANDING
BETWEEN
ERDA AND NRC

1. In order to support NRC's confirmatory assessment program, it is expected that NRC will reimburse ERDA for NRC's reactor safety research experimental and analytical programs carried out in the LOFT facility. ERDA will budget for operation of LOFT, as an NRC-reimbursable activity, except that ERDA will budget for such of its own work as may be carried out in LOFT, with NRC agreement. NRC and ERDA will coordinate their budget formulation activities in relation to LOFT requirements and will provide each other mutual support in budget jurtification and hearings before OMB and Congress.

2. LOFT is dedicated at this time to the NRC program; however, there may develop opportunities for its use in ERDA's reactor safety research program to which NRC and ERDA may jointly agree. NRC and ERDA will agree regarding the execution of NRC's experimental and analystical program at the LOFT facility. The NRC program will be carried out at the LOFT facility as outlined below.

   a. NRC will formulate with the ERDA contractor the experimental and analytical program (scope, schedule, and funding) including the quantitative and qualititative resources required to carry out the program. NRC will insure that ERDA is fully and currently informed of the status of program formulation.

   b. NRC will provide the proposed program scope, schedule, and funding level to ERDA.

   c. ERDA will determine whether the resources are adequate to carry out the program within defined scope and schedule and that the experiments are acceptable from a safety point of view.

   d. ERDA will assure that it has adequate resources to carry out any responsibilities it may have as provided below.

   e. ERDA will authorize the ERDA contractor to carry out the experimental and analytical program within the agreed upon scope, funding and schedule and under the program direction of NRC.

   f. Any changes to the NRC program (scope, schedule, and funding) will follow the procedures in a. through e. above.

   g. In the event of a desire by NRC or ERDA to undertake experiments that are planned to result in a major disabling of the facility to conduct further experiments, approval of the Commission itself and of the Administrator of ERDA will be required. The estimated cost and responsibility for funding of restoration of decommissioning of the facility to a suitable state will be agreed to at the time of such a decision.

   h. NRC and ERDA agree that program direction of the NRC experimental and analytical program at the LOFT facility is an NRC responsibility. In addition both parties agree that overall management of contractor operations at INEL, including assurance that work is performed in accordance with the terms of the contract, is an ERDA responsibility. Accordingly, both parties agree that close and continuous coordination between NRC and ERDA personnel is vital to successful program operations at INEL. To insure such coordination neither party will address directives to the contractor without due regard for the responsibilities and policies set forth in this Memorandum of Understanding. When a proposed directive to the contractor is considered by either party to result in an unacceptable compromise of either ERDA or NRC program and the matter cannot be resolved between INEL and NRC personnel, the matter will be referred promptly to ERDA and NRC Headquarters for decision.

   i. ERDA will be responsible for assuring that:

      (1) the work for NRC in the LOFT program does not unduly encumber requirements for ERDA programs at the INEL,

(2) the funding being provided by NRC for the LOFT program is adequate to cover the costs incurred and expected, including supporting facilities and site services,

(3) the requirements on the LOFT contractor call for a steady continuing schedule (avoiding job-shop requirements),

(4) the responsibilities of ERDA with respect to safety, safeguards, and physical security of the LOFT program are properly discharged including the issuance of necessary directives and authorizations,

(5) other potentially adverse effects on ERDA of conduct of the LOFT program are identified and action taken to avoid them,

(6) the information required by NRC to discharge its responsibilities is provided in a timely manner.

j. NRC will be responsible that its program direction to the contractor is consistent with the requirements that:

(1) the NRC LOFT program remains within the agreed-upon envelope of cost, schedule, and objectives; and that it is responsive to the NRC requirements,

(2) the resources provided for the conduct of the LOFT program are appropriate to the needs,

(3) the facilities provided by ERDA are suitable for the conduct of the agreed-upon research program,

(4) expenditures remain close the the anticipated and agreed-upon rates,

(5) the information required by ERDA to discharge its responsibilities is provided in a timely manner.

3. NRC may permit the participation of foreign countries in its experimental and analytical programs at LOFT subject to the conditions discussed below. NRC and ERDA will mutually agree in the formulation and execution of any agreements with foreign countries for such participation, including the stationing of foreign nationals at LOFT. ERDA will retain the right to disapprove any foreign involvements for reasons of national security.

4. ERDA will manage and direct the completion of LOFT, including the Mobile Test Assembly, to the point of readiness for the first nuclear loss-of-coolant experiment, providing, within Congressional and budgetary constraints, any funds necessary, in addition to those for which NRC has budgeted for LOFT construction in FY 1976, to complete LOFT on schedule in accordance with an approved baseline design requirement and a schedule to be jointly agreed between ERDA and NRC. NRC will provide to ERDA experimental requirements and guidance with respect to completion of this facility and will certify the facility's acceptability for NRC program requirements. Additional funding due to NRC requested changes in the LOFT design baseline will be reimbursed to ERDA by NRC.

5. Modifications to the LOFT that are of a major nature, i.e., involving new or significantly changed objectives to be achieved through the use of the test facility, will be budgeted for and managed by ERDA as in paragraph 4. above.

6. ERDA will budget for, manage and direct at LOFT supporting activities, general plant equipment, and engineering in support of major plant modifications.

7. All technical information and data associated with NRC work will be available to both agencies.

8. ERDA will provide for direct NRC personnel access to the ERDA facilities to enable them to fulfill their experimental program responsibilities.

9. NRC will reimburse ERDA for special items, including modifications, necessary to the continuing conduct of its experimental program (such as experimental instrumentation). The design and installation will be performed under NRC direction subject to ERDA concurrence in safety aspects. NRC and ERDA will agree on the scope, funding and schedule for such work to be performed by the Contractor and any direction by NRC to the Contractor outside the agreed upon scope, funding and schedule must be through the ERDA Contracting Officer.

10. ERDA will fund for any capital equipment essential to operation of the facility. Any capital equipment that is budgeted for by NRC for use in the conduct of its experimental and analytical programs will remain the exclusive property of NRC.

QUESTION 16. DOE is currently defining, with the help of the Commission and industry, a comprehensive program of reactor safety research. If they establish needs in areas currently covered by NRC research, will the NRC provide a schedule and NRC project manager responsible for satisfying the needs?

ANSWER.

As your question indicates, DOE, in response to PL 96-567, The Nuclear Safety Research, Development, and Demonstration Act of 1980, has convened a number of technical working groups in which NRC and industry respresentatives are participating to define needed safety research. If these groups establish needs in areas currently covered by NRC research, NRC will provide the schedule and access to the project manager. If a new issue is identified which the Commission feels requires resolution in order to protect the health and safety of the public and DOE is unable to fund the research, the need would undergo the prioritization embodied in the research endorsement process.

QUESTION 17. Do you feel that NRC should be "providing guidance" to DOE on waste form integrity, and other technical design parameters, or should they more properly be providing overall repository performance requirements and validating DOE data supporting their conformance to these requirements? Our concern is that NRC is unnecessarily involved in the design process in an area in which they will be making licensing decisions.

ANSWER.

The proposed NRC regulations for high level waste disposal provide for an overall repository performance requirement as well as performance objectives for major subsystems of the repository.

The intent is to set the requirements for a safe and licensable repository. In view of the technical uncertainties associated with demonstration of compliance with an overall system standard, the NRC considers that minimum performance requirements should also be met by the engineered and geologic subsystems. These are expressed as performance standards rather than detailed design parameters in order to provide to DOE the maximum possible latitude for design, consistent with requirements for a favorable licensing decision.

QUESTION 18. Has NRC conducted any research defining the impact on safety of mandating a high-point vent as a result of concerns raised by TMI? Has the safety improvement credited to this requirement been compared to safety degradation due to the additional penetration required?

ANSWER.

NRC has conducted calculations of expected plant response, and investigated the safety impact, of mandating a high-point vent in the reactor coolant system. The high-point vent is sized such that the rate of mass loss from the reactor coolant system will be less than the mass added by the charging pumps. Therefore, the pumping systems available in the power plant can provide more mass than will be removed through the high-point vent system.

The high-point vent system in all but a few cases has been connected to an existing penetration in the vessel. Therefore, no safety degradation results from requiring this system in terms of requiring additional penetrations. Also, the additional high pressure piping required to install the high-point vent system is very small and would not significantly increase the piping failure probability (the small break LOCA probability is now estimated to be about 1×10^{-2} per year). NRC believes that a safety improvement has occurred by adding the high-point vent system.

QUESTION 19. What problems would you foresee if industry were to be responsible for safety research?

ANSWER.

Assuming the NRC would continue to do sufficient research to keep its independent regulatory posture, the major problems which the staff would foresee if industry were responsible for safety research are:

1. Insufficient incentive to sponsor safety research in areas where industry perceives that there is a low probability of occurrence;
2. Achieving consolidation of effort to ensure a coordinated program and an equitable distribution of costs between utilities who are primarily interested in existing plant design and vendors who are primarily interested in future plant design.
3. Industry may focus on near-term problems and not develop long-range safety research programs.

Much of the reactor safety research is broad in scope, and beyond what might be appropriate to the means of individual utilities or vendors. Also, different segments of the industry respond to research needs according to their specific interests which may be quite narrow. EPRI, which is supported by the utility industry, provides an institutional approach which provides a broader base of support for reactor safety research. For this reason NRC has worked closely with EPRI to coordinate our safety research program both in cooperative projects and to avoid duplication. In an analogous context NRC works closely with INPO on operational problems because they also provide a broad base of institutional support in that area.

QUESTION 20. What research activities are planned in support of the trend towards standardization in design of nuclear power plants?

ANSWER.

The standardization in design of nuclear power plants is fundamentally a licensing issue; however, there is a role that research plays. Our research program on the qualification of electrical and mechanical equipment is expected to provide the necessary qualification procedures for a standardized design. These procedures will also assist in addressing the problem of replacement parts in standardized plants.

QUESTION 21. In addition to the requirements that NRC has imposed on access control for nuclear power plants, what is being done to prevent such facilities from sabotage by an insider? How do we compare in such protective measures with European countries?

ANSWER.

As the question indicates, current NRC regulations for defense-in-depth physical protection of plants and materials include significant requirements to control access in nuclear power plants.

These access control requirements, which include entry search requirements, badging and identification requirements, instrusion alarming, and tightly controlled access to vital area locations provide significant protection against unauthorized entry of either outside adversaries or insiders.

The Commission is considering establishing additional regulations for protecting facilities from sabotage by an insider. These regulations would establish screening requirements for individuals seeking unescorted access to protected and vital areas. The measures being considered include background investigations, psychological testing, and behavioral observation.

In addition to access control requirements, the NRC is funding research to develop design criteria for sabotage protection at future plants. Results of this research for new plants were reported in NUREG/CR 1345. Three follow-on research reports which discuss damage control measures and potential sabotage protection design changes for existing reactors have been completed by Sandia National Laboratories (SNL) and are in the process of being published. Additional research to better model system interactions and common mode failures which could result from acts of sabotage has recently been initiated. This projected work will also address the sabotage vulnerability of spatially extended systems and non-monitored standby components. The results of this reserach project, which will extend through FY85, will be used in the safeguards licensing process and to assess the effectiveness of current regulations to project against sabotage by an insider.

Based on information available through the DOE/NRC Foreign Physical Security Review visits, the staff believes that our measures to protect against sabotage by an insider generally exceed those employed by European countries.

The staff concludes that the access control requirements of West European countries are considerably less stringent than those imposed by 10 CFR 73.55. The requirements for security force qualification and training, in Appendix B 10 CFR Part 73, "Physical Protection of Plants and Materials," are also more stringent than the current practices of West European protective forces for power reactor sites. The NRC staff has not identified any methods for guarding against insider sabotage currently employed or proposed by European countries which have not been examined by the NRC staff or implemented in NRC regulations.

The European concept of bunkered engineered safety systems, which provide redundant safety features to protect against earthquakes, fires, and plane crashes, do provide some protection against insider sabotage.

QUESTION 22. How do the Commissioners supervise and direct the research efforts of their staff?

ANSWER.

The Commission provides guidance to all offices through:

1. Program Planning Guidance;
2. Designation of Unresolved Safety Issues;
3. Designation of Generic Issues;
4. Designation of TMI Action Items;
5. Requested rulemaking; and
6. Review and approval of the research budget.

The implementation of the research program supporting these policies is reviewed by the Commission in the Long Range Research Plan and the research budget. However, the principle direction of the research efforts is given by the Director of Research, following the general policy guidelines established by the Commission.

QUESTION 23. If additional research is needed for further revisions to Appendix K, do you propose to use the LOFT or other facilities for that research?

ANSWER.

No experiments are needed in the LOFT facility to support revisions to Appendix K in addition to those presently planned. The final LOFT experiment, which is scheduled to be conducted in February 1983, will be a simulation of a large break loss of coolant accident in a pressurized water reactor using end-of-life fuel conditions. This experiment has been designed to provide additional information useful for the assessment of the Appendix K fuel response models. Some cladding damage is predicted to occur during this experiment, which will provide additional results to verify experimental data already in hand.

In addition to the LOFT experiments, the 2D/3D and Semiscale programs will continue to provide experimental data to support a future Appendix K revision. These data will be used to assess the conservatism in the reflood, ECC bypass, and thermal hydraulic models required by Appendix K.

APPENDIX III

SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION

STAFF REPORT: INDUSTRY QUESTIONNAIRE ON NRC RESEARCH PROGRAM

The staff of the Subcommittee on Energy Research and Production conducted a survey of nuclear industry representatives in conjunction with hearings May 18 and 19 and June 15 on the Nuclear Regulatory Commission research program.

The respondants generally felt that the NRC research program has become more meaningful in the last few years. Nevertheless, the survey responses indicated general consensus on the following points:

-- NRC has mandated a number of "safety fixes" without proper assessment of their impact on overall plant safety;

-- Although the NRC program now covers the most important areas of safety research, it still lacks sufficient focus on key issues and often pursues research beyond the point of maximum benefit;

-- Cooperative research programs between NRC and industry produce results more efficiently and effectively than separate programs;

-- Industry should be encouraged to review NRC programs, especially during formative stages of new projects;

-- NRC research results which would reduce unnecessary conservatism are not adequately reflected in NRC regulations whereas findings which would increase conservatism are immediately applied, often on the basis of limited and inconclusive tests;

-- NRC needs a more systematic approach to establish research priorities;

-- Improved communication is still required between the research and licensing offices of the NRC.

Individual answers to the survey questions will not be published because respondants were promised anonymity. However, the attached material summarizes responses to each of the questions asked.

NRC RESEARCH
SUMMARY OF INDUSTRY QUESTIONNAIRE RESPONSES

QUESTION #1: Many people have criticized the NRC in that too many ad hoc "safety fixes" have been mandated without a proper assessment of their impact on the overall safety of power plants. Do you agree with this criticism? If yes, please provide any examples of safety fixes that may compromise safety and explain what the NRC should do in its research effort to help rectify their current approach.

SUMMARY OF RESPONSES: Nearly all respondants agreed with this criticism and numerous examples were cited. The responses, however, were primarily aimed not at innaccurate or misleading research results, but at lack of communication between the regulatory and research offices of the NRC. A number of the respondants were aware that the problem is recognized within the NRC; two quoted the NRC survey of utility management (SECY-81-437, July 22, 1981) which found that "the pace and nature of regulatory actions have created a potential safety problem of unknown dimensions." The creation of a Deputy Executive Director for Regional Operations and Generic Requirements was lauded by a number of respondants as a step in the right direction.

Among those "safety fixes" which respondants cited as not sufficiently researched for their overall impact on the public health and safety were reactor vessel head vents, water level indicators, some emergency procedures, initial primary coolant pump shutdown recommendations after TMI, seismic design for piping, and automatically closing fire dampers in ventilation systems.

Many respondants discussed general factors which they felt should be taken into account in future regulatory "safety fixes." Among these were the impact of design changes on plant complexity, judicious timing of requirements to minimize operator training deficiencies, and the uncertainty engendered by requiring state-of-the-art technology without sufficient testing.

Additional steps which respondants felt could improve NRC performance in the future are increased use of probabilistic risk assessment techniques to investigate competing risks, development of improved plant system analysis tools, increased communication between the regulatory offices and safety researchers both at NRC and in industry, and industry input and peer review during the scoping and justification of research programs as well as at the completion.

QUESTION #2: In your estimation, does the NRC research program properly focus its efforts on the most important issues for overall safety? If not, please explain. Do you feel that the NRC research program is responsive to the NRC mission to protect the public health and safety and the common defense and security in the civilian uses of nuclear energy?

SUMMARY OF RESPONSES: This question evoked a less uniform response than Question #1. However, a number of predominant themes were expressed,

Many of the respondants noted that although the NRC in the past had neglected some important safety issues, they have recently been moving to rectify this situation. In general, the respondants indicated that NRC currently covers the most important areas for safety research. Considerable concern was expressed, however, about the focus of NRC research programs and their relationship to key safety issues. Respondants stated that NRC often pursues its research programs beyond the point of maximum benefit, consuming resources that could be more productively applied to other issues. Inadequate communication between regulatory and research offices, and undue emphasis on extremely unlikely accident scenarios were other concerns noted. Nearly every response that mentioned specific issues focused on the need for definitive information about the release and transport of fission products during severe accidents.

QUESTION #3: Do you feel that there is sufficient interaction between the NRC research staff and industry to allow the staff to make informed trade-offs between design and operating constraints? If not, how could this interface be improved?

SUMMARY OF RESPONSES: Respondants generally indicated that current interactions between NRC and industry are inadequate, but that the situation has improved markedly in recent years. A number pointed to the industry IDCOR program as an outstanding example. The adversarial relationship which had existed in the past was criticized as counterproductive and inefficient. Increased cooperation with industry was seen as one way to conserve resources by reducing duplicative research and sharpening the focus of NRC research.

Lack of adequate communication between regulatory and research offices within NRC was again noted in a number of responses.

Many of the respondants recommended more widespread use of advisory groups, particularly in early phases of new projects. It was suggested that industry could assist the NRC in designing programs to yield results that are more directly applicable. It was also suggested that NRC seek industry recommendations on the scope of current and planned R&D programs. These steps were seen as ways to increase the incentive for NRC to resolve safety issues as quickly as possible and to terminate associated research programs in a more timely manner. One respondant remarked that high-level agreements between NRC and industry groups such as EPRI and INPO don't seem to filter down to the staff level.

QUESTION #4: Do you feel that NRC research results are adequately reflected in the licensing activities of the NRC? If not, please cite counter examples.

SUMMARY OF RESPONSES: This question evoked unanimous and emphatic responses to the effect that research results are inadequately reflected in NRC regulations, particularly when they would result in relaxed requirements. Many respondants noted a marked contrast between findings implying a need for increased regulatory conservatism which are immediately acted upon even on the basis of limited and inconclusive results, and findings demonstrating unnecessary conservatism which are subjected to exhaustive and interminable research programs before having any impact. An example offered of the former were post-TMI requirements which were mandated prior to research. In some cases subsequent research has shown them to be ineffective or counter to overall safety. The two most heavily used examples of accepted results not reflected in the regulatory process were the LOFT tests on Emergency Core Cooling Systems, which have not been reflected in 10 CFR 50 Appendix K, and the indications on iodine release and transport at TMI. Many respondants favored an acceleration of current research on fission product release and transport, and possibly a conservative but more realistic interim set of assumptions.

Some reponses implied that research results are only used when they support models already accepted by the regulatory offices.

One respondant provided an interesting and thoughtful response which noted a fundamental incompatibility of format between current regulatory practice and research results. He described the regulations as formalistic requirements which by their very nature fragment the plant into subsystems, whereas many research programs deal by necessity with complex interactions among subsystems having an integrated effect on the overall plant. This respondant was concerned with the loss of valuable information when research results are cast in the current regulatory formats.

QUESTION #5: Are NRC research results useful to industry in designing and operating reactors? Are the results readily available to industry in useable form?

SUMMARY OF RESPONSES: The consensus among respondants is that research results are generally available to industry but only marginally useful. Computer codes were cited as benchmarks for industry codes but not replacements. A few respondants noted a fundamental difference in perspective, with NRC concentrating on worst-case accidents and industry emphasizing day-to-day operations and more probable accident sequences. Some of those who responded with examples of research useful to industry recommended more emphasis on proper documentation and quantitative analysis.

QUESTION #6: What should be the priorities of the DOE safety research program? What can be done to improve the interface between NRC and DOE?

SUMMARY OF RESPONSES: Responses demonstrated more differences among respondants than answers to any other question. Among those roles envisioned for DOE, the most prevalent were development of advanced reactors, conduct of comparative safety studies of alternative technologies for electric energy generation, and coordination of interactions between NRC and industry researchers. With the exception of one respondant who felt that interation between NRC and DOE is immaterial given their respective budgets, there was a general consensus that the interface between the two should be clarified. However, various suggestions included a definitive line drawn between the two agencies, a major focus for each with overlapping responsibilities, or a single agency (most logically NRC) in charge of safety research with the other (DOE) exercising oversight and review.

Some respondants felt that DOE could develop a better focus on overall safety because it is unimpeded by the specific regulatory structure of the NRC. DOE was complimented on the formation of working groups pursuant to P.L. 96-567 as a potentially effective method of establishing safety priorities, but some respondants called for stronger management of the program. The recommendation most common to all responses was the establishment of a systematic approach to setting priorities for a national reactor safety program including all safety research participants.

APPENDIX IV

SUMMARY

SUBCOMMITTEE ON ENERGY RESEARCH AND PRODUCTION

PRELIMINARY FINDINGS AND RECOMMENDATIONS ON RESEARCH PROGRAMS CONDUCTED BY THE NUCLEAR REGULATORY COMMISSION

General

The Subcommittee on Energy Research and Production held hearings on May 18 and 19 on the research program conducted by the Nuclear Regulatory Commission (NRC).

Those NRC research programs examined by the Subcommittee staff were found to be generally of a high technical caliber. The hearings, however, uncovered serious concern about the management of NRC research, particularly in regard to the establishment of research priorities and the definition of technical objectives. As described in the findings below, the major problems are rooted in the organizational structure of the Commission, which doesn't clearly assign responsibility for defining programs and allocating research resources to those people who have the information and expertise to assess the costs and potential benefits of research projects. In a number of the findings, such as the relationship of the Office of Nuclear Regulatory Research to industry, the Commission is viewed as aware of the problems and moving in the right direction but requiring substantial additional improvement.

Findings and Recommendations

1. FINDING: RESULTS OF RESEARCH DIRECTED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH ARE NOT ADEQUATELY REFLECTED IN THE RULES AND REGULATIONS OF THE NRC.

   RECOMMENDATIONS:

   1a. Update 10 CFR 50, Appendix K, which covers emergency core cooling system evaluation models.

1b. Review NRC research programs and list potential regulatory impacts.

2. FINDING: THE DISTINCTION HISTORICALLY DRAWN BETWEEN "CONFIRMATORY" AND "SAFETY" RESEARCH AT NRC IS ARTIFICIAL AND COUNTER-PRODUCTIVE.

   RECOMMENDATIONS:

   2a. Reword the charter of the Office of Nuclear Regulatory Research to support more broadly the overall mission of the NRC.

   2b. Expand research on the impact of regulatory formats and processes on assured and improved safety.

3. FINDING: THE SETTING OF RESEARCH PRIORITIES AT THE NRC IS AN AD-HOC PROCESS WITH NO OVERALL FRAMEWORK OR SYSTEMS OVERVIEW AND VERY LITTLE DELINEATION OF POTENTIAL COSTS AND BENEFITS FOR INDIVIDUAL PROGRAMS.

   RECOMMENDATIONS:

   3a. Restructure the Long Range Research Plan to:

      - Include introductory material describing a logical framework relating research projects to overall safety.
      - Reorganize program-specific detail to fit this framework.
      - Clearly delineate areas where future research is needed, even if current budget limitations preclude initiating projects in the upcoming fiscal year.
      - Explicitly define deliverables to be provided to user offices.
      - Include in the plan estimates of necessary resources and recommended schedules, even if these have to be heavily qualified (no penalties should be applied for reassessing these schedules and resource estimates as programs become better understood).

3b. Submit to Congress with the budget request for each fiscal year a description of deliverables, major intermediate milestones, total estimated resource requirements and potential impact on safety for each program.

3c. Develop a systems engineering organization to formulate an overview of factors affecting plant safety and to set research priorities and objectives accordingly.

4. FINDING: NRC'S DOMINANT ROLE IN U.S. SAFETY RESEARCH TENDS TO DIRECT LICENSEE RESEARCH RESOURCES TOWARDS ASSURING COMPLIANCE RATHER THAN IMPROVING PLANT SAFETY.

RECOMMENDATIONS:

4a. Place the burden of reactor safety on the licensee by requiring more safety research of the industrial community and de-emphasizing NRC's predominant research role.

5. FINDING: DESPITE A STRONG RECENT TREND TOWARD INCREASING COOPERATION BETWEEN THE OFFICE OF NUCLEAR REGULATORY RESEARCH AND THE INDUSTRIAL COMMUNITY, THE CURRENT INTERFACE IS NOT ADEQUATE FOR NRC RESEARCHERS TO INCORPORATE MAINTENANCE AND OPERATIONS CONSIDERATIONS INTO THEIR SAFETY RESEARCH.

RECOMMENDATIONS:

5a. Continue to increase substantive cooperative programs with the Electric Power Research Institute (EPRI) and the Institute of Nuclear Power Operations (INPO).

5b. Request EPRI or INPO review at the initiation or completion of any NRC research program involving operational procedures or reactions either as an input or as a deliverable product.

5c. Formally invite EPRI comments on the NRC Research Budget and on the Long Range Research Plan when these documents are reviewed by the Advisory Committee on Reactor Safeguards (ACRS).

6. FINDING: THE INTERFACE BETWEEN THE OFFICE OF NUCLEAR REGULATORY RESEARCH AND THE DEPARTMENT OF ENERGY (DOE) IS NEITHER CLEARLY DEFINED NOR WELL UNDERSTOOD.

RECOMMENDATIONS:

6a. Establish with DOE a Cooperative Review Committee to coordinate research of mutual interest and to recommend regulatory changes to reflect research results.

6b. Negotiate specific memoranda of understanding with the Department of Energy to cover LWR Safety Research, Advanced Reactor Safety Research and Research on Safe Management of Nuclear Waste.

6c. Review all current research at NRC and DOE on Advanced Reactors and Waste Management. Coordinate the respective programs of the two organizations to eliminate any unnecessary duplication of effort.

6d. Continue cooperating with the Department of Energy as they define a Comprehensive Reactor Safety Research Program pursuant to The Nuclear Safety Research Development and Demonstration Act of 1980 (P.L. 96-567).

7. FINDING: RESEARCH ACTIVITIES AT NRC NOT DIRECTED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH ARE NOT ADEQUATELY INTEGRATED WITH THE OVERALL RESEARCH PROGRAM.

RECOMMENDATIONS:

7a. Include all research directed by NRC in the Long Range Research Plan and other planning documents so that relationships among programs are clearly defined.

APPENDIX V

Nuclear Safety Oversight Committee

REPORT OF

THE REACTOR SAFETY RESEARCH REVIEW GROUP

Norman C. Rasmussen, Chairman
Herbert J. C. Kouts, Vice Chairman

Spencer H. Bush
Thomas J. Connolly
Herbert G. MacPherson
David Okrent
Lombard Squires
Edwin L. Zebroski

SUBMITTED TO

THE PRESIDENT'S NUCLEAR SAFETY OVERSIGHT COMMITTEE

Governor Bruce Babbitt, Chairman

John Deutch
Marvin Goldberger
Harold Lewis

Steven Ebbin, Executive Director

SEPTEMBER 1981

September 1981

The Honorable Bruce Babbitt
Chairman
Nuclear Safety Oversight Committee
Washington, D.C. 20545

Dear Gov. Babbitt:

I am pleased to transmit herewith the report of the Reactor Safety Research Review Group. You will note that it contains a number of recommendations directed toward improving the national reactor safety research program, with particular attention to the Nuclear Regulatory Commission research program.

The text contains a number of recommendations considered important enough to be singled out. Of these, 26 were considered to be of a higher level of urgency and these have been listed at the front of the report for emphasis.

We hope these recommendations will be of use to your committee in carrying out its responsibilities. I would like to take this opportunity to thank you for your support and that of your staff, especially Mr. Steven Ebbin, in helping us carry out this review.

I look forward to appearing before your committee on September 23, 1981 to discuss this report in person.

Very truly yours,

Norman C Rasmussen

Norman C. Rasmussen

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Introduction

Scope of Review

This review of the national program of reactor safety research was performed at the request of the Nuclear Safety Oversight Committee. It has been limited to research related to safety of nuclear power plants, and specifically does not include other aspects of the nuclear fuel cycle. It has been further restricted to safety of light water moderated and cooled nuclear power plants of the types used in the United States.

The Review Group

To conduct the review, NSOC selected a group of individuals with extensive experience in reactor safety issues. Professor Norman C. Rasmussen of the Department of Nuclear Engineering at the Massachusetts Institute of Technology was asked to chair this Review Group. The other members were:

Dr. Spencer Bush	Battelle Pacific Northwest Laboratories
Dr. Thomas Connolly	Stanford University
Dr. Herbert J.C. Kouts	Brookhaven National Laboratory
Dr. Herbert G. MacPherson	Institute for Energy Analysis
Dr. David Okrent	University of California at Los Angeles
Mr. Lombard Squires (retired)	E.I. du Pont de Nemours Company
Dr. Edwin Zebroski	Nuclear Safety Analysis Center

Performance of the Review

In performing its review, the Group had the benefit of meetings with a number of members of the U.S. water reactor safety research community. An organizational meeting was held on April 24, 1981, in Chicago. A second meeting was held on June 4-5 in Washington,D.C., with representatives of NRC's Office of Nuclear Regulatory Research and representatives of DOE. On June 25-26, a meeting was held in Denver with representatives of the four LWR vendors (Westinghouse, Combustion Engineering, Babcock and Wilcox, and General Electric), selected utilities (Commonwealth Edison, Tennessee Valley Authority, Duke Power, and Pacific Gas and Electric), and individuals representing consulting companies (Saul Levine of NUS, Robert Budnitz of Teknekron, and Mario Fontana representing IDCOR*). In addition, individual members of the Review Group visited all of the National Laboratories where the bulk of the nation's safety research is done. Discussions were held there with staff members at both the managerial and working level. Comments and views expressed to the Review Group at both individual and group meetings have been factored into this report, but without attribution.

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* Industry Degraded Core Rulemaking Program

An extended meeting of the Review Group was held in Seattle from August 10-14, 1981. A final meeting was held in Washington on September 2, 1981.

Research Considered in the Review

Research on light water reactor safety is sponsored by many organizations. These include the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) in the public sector, and reactor vendors and the utilities through owners' groups and the Electric Power Research Institute (EPRI) in the private sector. There is also a substantial and growing amount of water reactor safety research abroad, principally in the Federal Republic of Germany, France, Japan, and Sweden. Because many projects abroad are sponsored jointly with organizations in the U.S., and because the information developed in most national programs is fully exchanged, water reactor safety research should be viewed as an international effort. Some attention is therefore given here to contributions by other countries.

At the present time, more than half of the research on water reactor safety in the United States is sponsored by the NRC. In FY-82, for example, the budget of the Office of Nuclear Regulatory Research of NRC is $230 million, of which approximately $200 million is destined for light water reactor safety research. The research by DOE on this subject is very limited. Public Law 96-567 now calls for a substantial DOE role in this area. Implementation of this law is still under review, and budgetary allocations for it have yet to be made.

In the private sector, water reactor safety research is frequently sponsored by groups of utilities and/or nuclear steam supply vendors with specific common problems. The Electric Power Research Institute (EPRI), the research arm of the electric utilities, is also a sponsor of substantial research on light water reactor safety and reliability. In FY-82, EPRI's expenditures in this area are expected to amount to about $65 million. Since the accident at Three Mile Island, the industry has also formed the Nuclear Safety Analysis Center (NSAC) and the Institute for Nuclear Power Operations (INPO). These organizations sponsor additional activities to improve safety of light water reactors.

These estimates of $200 million expended annually in the public sector on light water reactor safety research, and perhaps half that value sponsored by the private sector, are necessarily arbitrary. Typical questions that arise in considering whether a given project by industry should be included are: is it research or is it simply engineering analysis, and is it research to improve reactor safety or to improve a vendor's product? For example, the question of whether a given research project is or is not safety research arises in connection with certain programs sponsored by owners' groups, which deal with product reliability (e.g., improvement of design of steam generators for PWRs). It is acceptable to argue that increased reliability leads to increased safety, but these programs are driven more by economic considerations than by safety considerations.

The NRC, on the other hand, also conducts many activities resembling research in its Technical Assistance Programs. These are not usually defined as research, even though some programs under the rubric of technical assistance have been long-term and have been exploratory in nature. These technical assistance programs have not been reviewed here.

Structure of the Report

The review that follows has been structured along the lines of accident analysis, because this brings up the safety issues in a natural way. The first section discusses research that helps to avoid accidents and research on the modes of accident initiation. This is followed by a section on research pertinent to the response of the plant to initiating events, including functioning of safety systems. The third includes research on the consequences within the plant if the protective systems were to fail; this includes fuel failure and liberation of fission products from fuel, transport of the fission products, and their release from the containment. The fourth section includes research on consequences of fission product release. The fifth includes questions which pervade the entire accident analysis. The sixth contains observations on organizational and institutional subjects.

Interspersed in the text of the report are a number of recommendations that the Review Group feels would improve the overall effectiveness of the national program. The most important of these are singled out in the text. Forty-two recommendations are emphasized in this way.

Conclusions and Recommendations

General Conclusion

The national research program on safety of light water reactors began to assume its present form and objectives only about eight years ago. The structure it assumed then was a reaction to questions raised during the long hearing of 1972-73 on criteria for effectiveness of emergency core cooling systems. The program has kept this initial orientation ever since, with little redirection to topics not originally included. This is understandable in the context of the length of time that has been found necessary to mount such a complex research program and to carry it to useful conclusions. In the intervening period, the program has made significant progress in answering questions on reactor safety. The NRC, its contractors, and those in industry who have joined in this program have earned and well deserve a substantial measure of credit for important contributions to the safety of the public.

The Large LOCA Program

In particular, the research to improve understanding of the large loss-of-coolant accident has been an unquestioned success. It has progressed to the point where the strong emphasis on this research objective can now be wound down. The LOFT program, which is the centerpiece of large LOCA research, has succeeded in its original objectives, and the Review Group agrees with views expressed elsewhere that moves to decommission LOFT can now be taken. In subsequent sections of this report, we identify high priority areas in which research results are urgently needed, and to which the resources saved through the phasing out of LOFT should be applied. Our recommendation is:

> The LOFT program should be phased out in an orderly manner. This Review Group is divided on whether the experimental program for LOFT should be completed in FY-82 or FY-83. (P.II-2)*

New Research

Very important issues have arisen in other areas in the past few years, and a number of these now compete for attention. Research programs have been started on some of these issues. In some other instances, though, research programs are required but do not exist. We have singled out a number of issues that urgently need research but on which no significant research is being done. Recommendations on these are given in the text of our report. The most important are listed below, in order of urgency and importance.

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* Each recommendation repeated in this section is accompanied by the page number of the main report where it may be found.

> A major effort should be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems, for both the reactor vessel and the containment. (p. II-3)
>
> A program should be initiated to develop and evaluate methods of providing more reliable electric station power, both AC and DC. Such a program would have significant value and an excellent cost-benefit ratio. (p. I-7)
>
> A program should be undertaken to examine the relative safety merits of rigid versus flexible seismic design of piping. The program should consider allowance for inelastic behavior of piping and evaluate the potentially adverse impact of an excessive number of hangers and snubbers on in-service inspection and the possibility of crack initiation during normal operation due to snubber malfunction. (p. I-5)
>
> Industry or DOE programs should be established to develop valves with more reliability of function under normal and abnormal operating conditions. (p. I-4)
>
> DOE and industry should now proceed with the long-planned, comprehensive program to develop and demonstrate primary system decontamination techniques applicable to LWRs. This program should include techniques which reduce the initial deposition of radioactive materials. (p. I-5)
>
> A national LWR system simulation program should be undertaken cooperatively by DOE and NPC. This program should treat both PWRs and BWRs generically. The principal goal should be the development of computational capability to study LWR behavior in real time or faster through a wide range of severe transients including accidents involving extensive core damage. (p. II-4)
>
> NRC programs should be established to determine the mode and likelihood of severe damage by water hammer, to permit engineering solution to prevention of damage to the plant and especially to safety-related systems. (p. I-4)
>
> The NRC should undertake the studies needed to develop design measures and/or other measures which protect against sabotage by an insider while not compromising safety in other respects. (p. I-7)

Existing Research

Among the research programs now being supported, several have received specific attention in our report. In most cases, this is because either increases or decreases in emphasis are appropriate. In a few cases we have said that current levels of support are reasonable. Among our recommendations on existing research programs, the following are especially important. They are given in order of urgency and importance:*

> Probabilistic risk analysis indicates that about half the risk from reactor accidents is attributable to human error. The body of knowledge concerning pertinent human factors is inadequate, and it is important that further research be done in this area to provide an adequate technical basis for regulatory activities. The NRC should establish relative priorities, so that its research program will be structured to obtain the most important information first. The important information is that which can be used to improve control room design, operator aids, and selection and training of operators, which should lead to reduction of rates of human error. Further, the NRC program should be coordinated with industry efforts. (p. V-2)
>
> The NRC, DOE, and industry have ongoing programs to provide improved understanding of radioactive releases possible after severe reactor accidents. These should encompass a range of conditions and accident scenarios sufficient for broad understanding of the processes of transport and release of the important radioactive isotopes. The NRC program should provide as much information as possible for the degraded core cooling hearings. There is also a need for a long-range program in this area. (p. III-7)
>
> The NRC program on the small loss-of-coolant accident should be continued, to the point where capability to answer important questions of thermal-hydraulics and fuel performance is assured. (p. II-3)
>
> The possibility of cold repressurization of reactor pressure vessels that have undergone large shifts in nil ductility transition temperature provides a potential mechanism for a major failure of the pressure boundary. The substantial programs underway or in the advanced planning stage should be funded and coordinated to assure that the issue is resolved in a 1-3 year time frame. (p. I-3)
>
> Ample federal funds should be made available for developing timely information through examining the degraded core in TMI-2. The current program of DOE to carry out such investigations should be supported. (p. III-4)

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* Dr. Okrent has the following additional comment:
All of the above recommendations are important. If required to assign priorities, I would arrive at a somewhat different order. I would place the 4th, 9th, and 10th recommendations on cold repressurization, minimum engineered safety features, and the degraded core cooling rulemaking, respectively, in a group at the top of the list. I would place the 1st, 2nd, and 11th on human factors, the radioactive source term, and mitigation features, respectively, in the next highest group. I would place the 3rd, 7th, and 8th on the small LOCA, probabilistic analysis, and control systems, respectively, in a third group, while the 5th and 6th on the TMI-2 core and hydrogen, though still important, would be in a last group.

> The NRC is initiating a research program on the issue of evolution and burning of hydrogen in accidents. This program is important and should be pursued expeditiously. Complementary programs by industry should be taken into account in structuring the NRC's research program. The significance of containment design and volume should also be taken into account. (p. III-9)
>
> The use of probabilistic analysis is rapidly expanding. A significantly larger research program is needed to improve and standardize the methodology as much as possible. Particular areas that need improvement are: methods of handling common cause failures, data collection and analysis, accuracy of the estimated frequency of very low probability events, and assurance of the quality of analyses. (p. V-4)
>
> Research on the role and importance of plant control systems in LWR safety should be greatly augmented. (p. I-7)
>
> The NRC should promptly identify the information needs of the rulemaking on minimum engineered safety features of future LWRs that can be supplied in time by safety research, and assign the necessary priority and resources to this task. (p. II-5)
>
> The NRC should promptly identify that information which it needs for the degraded core cooling rulemaking and which safety research can supply in time for the hearing. It should assign the priority and resources to get the job done, deferring, if necessary, other less urgent degraded core safety research. (p. III-4)
>
> The NRC safety research program on mitigation of degraded core and core melt accidents should be modified as necessary to provide the information needed on alternative containment design concepts under consideration, including improved containment cooling, containment venting, venting and filtering, core debris retention, and hydrogen control. (p. III-10)

General Comments on NRC Research Programs

We also reviewed such pervasive matters as the logic and structure of NRC's safety research program, the way the benefits of research are incorporated (or not incorporated) into regulations, and the general relationship between the Office of Nuclear Regulatory Research and the other components of NRC. Among the observations we transmit are several recommendations on improvements to the way NRC plans, manages, and uses its research programs. The most important are:

> The Commission should encourage and promote the visible integration of the results of safety research into the regulatory process. Regulatory requirements should not rest on bad science when good science has become available. If retention of conservatism is desired because of uncertainties or for other reasons, this should be done through the application of explicit safety factors to a calculation based on best available methods, in accordance with good engineering practice. (p. VI-3)

> The limitation of NRC's programs to "confirmatory research" should be removed, so that exploratory research and research to improve safety can be undertaken when this looks like the better course to follow. (p. VI-3)
>
> NRC's Long Range Research Plan (NUREG-0740) should be restructured to follow lines of Agency objectives. This should infuse it with a more up-to-date logical structure and discourage the tendency to support more of what has already been done. Introduction and continuation of guidance at the Commission level will be of fundamental importance in confirming the objectives and restructuring the plan. The restructured long-range plan should be more specific in defining the deliverables in each program. (p. VI-3)
>
> A policy should be instituted whereby probabilistic risk assessment is used as one tool in establishing priorities of research programs. (p. VI-6)

The last of the above recommendations should be read in a context broader than NRC programs on safety research. It is good advice for all applied research.

Research by DOE

In Public Law 96-567, Congress directed the Department of Energy to undertake studies leading to a program to contribute to sound design and safety of nuclear power plants. We have reviewed the DOE's reactions to these instructions and find them to be weak and generally unresponsive. We have two important recommendations on this subject, listed below:

> DOE should form a strong staff of technical and managerial personnel knowledgeable and experienced in the subject of water reactor safety, to develop and implement programs of safety research in this field. (p. VI-5)
>
> DOE should develop a program of generic research to improve water reactor safety, and assume a substantial responsibility in the area of accident prevention to supplement other programs designed to reduce the likelihood of nuclear power plant accidents. (p. VI-5)

I. Potential Accident Initiators and Their Prevention

For many years it has been well known that the occurrence of certain incidents during the operation of a nuclear power plant could start a chain of events leading to fuel damage and accidental release of radioactivity, if not responded to correctly. It is convenient to classify such events into those having external causes and those resulting from failures inside the plant. Risk analyses generally conclude that among the external causes, large earthquakes are the most likely to cause a serious accident. Other causes considered in safety analyses are floods, wind storms, external fires or explosions, and sometimes airplane crashes. Among the internal events considered are failures in the primary system boundary, anticipated or unanticipated transients requiring shutdown of an operating plant, and human errors. (Human errors are discussed in Section V.)

A. Internal Initiators

The main cooling system of a light water reactor (LWR) contains water at high temperature and pressure. If the pressure system boundary should leak or fail, water and steam would be rapidly ejected and a loss-of-coolant accident would be initiated. The outcome of such an event would depend upon the effectiveness of the emergency safety features of the plant in coping with the event.

Another broad class of initiating events is termed transients. These are events that would require plant shutdown, and thus the plant shutdown system must operate effectively and then the decay heat removal system must dissipate the heat generated by the radioactive decay of the fission products. Both classes of events are considered here.

1. The Primary Pressure Boundary

a. General Considerations

The reactor pressure boundaries of LWRs have been a major subject of attention. Most safety-related R&D programs emphasize pressure boundary integrity, possible degradation mechanisms, nondestructive examination (NDE) techniques for inspecting the pressure boundary, and loads, particularly under faulted conditions. Programs on functional performance of pumps and valves under extreme conditions have been relatively limited and deserve more attention. The areas of mechanical components and structural safety have substantial overlaps with regard to seismicity, load combinations, codes and standards, and behavior under transient and accident loads other than seismic; the programs are well conceived and have been meeting their goals.

The NRC fracture mechanics program needs to be examined in the context of national and international work in linear-elastic fracture mechanics (LEFM), elastic-plastic fracture mechanics (EPFM), and general yield fracture mechanics. Taken separately, the NRC program alone is considered inadequate; however, the overall coverage nationally and internationally is excellent. LEFM has been applied directly to nuclear systems for some time. It is recognized that there is substantial overlap among programs on EPFM, but this overlap is considered acceptable in such a relatively new field. The progress in EPFM has been rapid, and this methodology should be ready for application within a year. The ultimate test will be found in application to real systems.

Intergranular stress corrosion cracking (IGSCC) in austenitic stainless steels has been a major problem for BWRs and it has occurred in some PWR piping to a limited degree. Extensive research on this topic has been funded by NRC, EPRI, and industry (e.g., the BWR Owners Group). The phenomenology of IGSCC is now fairly well understood as are materials optimized to eliminate IGSCC in new piping systems. An area that has also been investigated extensively has to do with actions that might be taken to minimize IGSCC in existing systems. These actions include rigorous control of water chemistry, and means to reduce stress levels in sensitized regions (near welds). An example is the application of induction-heating-stress-relief (IHSR) pioneered in Japan, to place the inner surfaces in the heat-affected zone in compression. Despite extensive research and actual plant applications in Japan, there is still some question by the NRC staff regarding the degree of stress reversal attained. Some programs to further quantify the value of the process and to qualify it could be of value.

The steam generator tube degradation/integrity work is a part of a larger overall program funded nationally and internationally. The industry research program has led to a reduction in the rate of tube damage and required tube plugging. Several reactors have units with degraded steam generator tubing. These units will probably continue to operate until the number of tubes plugged is excessive. One area that has not been considered sufficiently using recent accident analysis codes is estimation of the consequences of a transient or some other failure that might lead in turn to the failure of a significant number of tubes. Such failures could lead to the degradation of ECCS function.

- Programs should be funded to apply the best available analytical methods to the assessment of effects of failure of degraded steam generator tubing on reactor transients and accidents.

The final area relevant to the pressure boundary is nondestructive evaluation. The extensive programs need to be examined in the context of other national and international programs. Programs are generally very good, with good rapport among the NDE community. The work seems to be progressing well toward adequate resolution of all issues. Adequate methods are becoming available, but extensive validation campaigns are needed to get code approvals and/or to support NRC acceptance while the code approval process is carried through.

b. Pressure Vessel Rupture

Analyses some years ago by AEC and ACRS resulted in the conclusion that the probability of reactor pressure vessel rupture was acceptably low. Recent reactor occurrences have, however, drawn attention to a possible RPV failure mechanism, not analyzed in the past, which is called cold repressurization.

Significant risk from cold repressurization in reactor pressure vessels could appear near end of life for some vessels. The initiating incident might be a partial blowdown of the PWR secondary side. If there were a flaw in the reactor pressure vessel beltline near the inner surface, if the vessel had undergone severe radiation damage in its beltline, and if the cooling rates were sufficiently rapid to generate high tensile loads around the flaw, fracture might in principle occur. Major programs are planned or are underway by NRC and industry. One area of possible inadequacy is in the ability of NDE to detect flaws near the inner surface. Much of the development of ultrasonic systems optimized for detection in this region has occurred in Europe. The problem is one of quantification of reliability and validation for code approval rather than a feeling that no UT systems exist capable of reliable detection.

- The possibility of cold repressurization of reactor pressure vessels that have undergone large shifts in nil ductility transition temperature provides a potential mechanism for a major failure of the pressure boundary. The substantial programs underway or in the advanced planning stage should be funded and coordinated to assure that the issue is resolved in a 1-3 year time frame.

c. Effects of Water Hammer

An issue of continuing importance is the effect of water hammer and water slugging on PWR and BWR piping systems. Water hammer incidents which have caused damage to piping, valves, valve operators, etc. have been extensively reviewed; however, research aimed at assessment of the severity of water hammer required for damage, and corrective measures, has been quite limited. Little is known concerning the probability of initiating a LOCA by water hammer.

- NRC programs should be established to determine the mode and likelihood of severe damage by water hammer, to permit engineering solution to prevention of damage to the plant and especially safety-related systems.

d. Valves

Valves represent a major possible source for the initiation of accidents, not so much by structural failure, but through loss of function. Definitive studies have been conducted on modes by which valves fail; however, little has been done to develop measures to minimize functional failure. Corrective action will require definitive research programs and close rapport with valve manufacturers. Establishment of this rapport is recognized as very difficult but necessary.

- Industry or DOE programs should be established to develop valves with more reliability of function under normal and abnormal operating conditions.

e. Pipe Design for Seismic Loads

A common complaint from utilities is the adverse effect that conservatisms in seismic design have had on piping systems. Conservatism in damping factors used to calculate response of piping to earthquakes has initiated a chain reaction adversely affecting maintenance, in-service inspection, and general piping reliability. Current regulations result in such a large number of supports and snubbers to comply with postulated seismic loads that it is difficult for maintenance or in-service inspection personnel to obtain access to piping. Furthermore, the rigidity of the systems leaves little margin for errors in installation. The NRC Seismic Safety Margins Research Program (SSMRP) has as one objective the assessment of behavior of piping under seismic loads. Previous programs have confirmed that large damping factors exist under loads, and have also developed inelastic response computer codes. These programs indicate the advantages of basing design on meaningful inelastic response rather than on conservative elastic response. Available records on petrochemical plants confirm that seismic loads rarely, if ever, have failed piping of diameter greater than 4 inches when such systems are left flexible and where little attention has been paid to seismic design. Increased flexibility in nuclear service should result in increased overall safety through better maintenance and in-service inspection and through design to conditions more forgiving under stress.

- A program should be undertaken to examine the relative safety merits of rigid versus flexible seismic design of piping. The program should consider allowance for inelastic behavior of piping and evaluate the potentially adverse impact of an excessive number of hangers and snubbers on in-service inspection and the possibility of crack initiation during normal operation due to snubber malfunction.

f. Decontamination of the Primary System

The primary system of an LWR becomes contaminated in service by the deposition of films of corrosion products consisting predominantly of iron and nickel hydrous oxides, with an appreciable amount of cobalt as the most important radionuclide. During sustained operation of the reactor, a progressive and irreversible increase in the level of radioactivity in the primary piping and equipment occurs, which prevents or inhibits "hands-on" access to these components after the reactor is shut down. Substantial research and development have been undertaken by industry over the years to understand the chemistry and to remove (decontaminate) the accumulated deposits.

An integrated research and development program is needed to establish techniques and procedures for routine on-line decontamination of commercial power reactors.

If such demonstrated techniques could be used industrywide, there would be fewer plant outages, better equipment surveillance and maintenance, enhanced safety of the primary system, and a major and urgently needed reduction in exposure of plant personnel.

- DOE and industry should now proceed with the long-planned, comprehensive program to develop and demonstrate primary system decontamination techniques applicable to LWRs. This program should include techniques which reduce the initial deposition of radioactive materials.

An important large pilot demonstration to decontaminate the primary system of Dresden-1 has been delayed several years by regulatory problems. These should be resolved promptly by NRC. DOE should give high priority to this demonstration and support the follow-on, full-scale decontamination of Quad City-1 using the dilute solution approach to be demonstrated at Dresden-1.

- The proposed decontamination demonstration at Dresden-1 should be undertaken expeditiously.

Because of a difference in reactor water chemistry, a low concentration process is more applicable to BWRs. Significant and encouraging progress has been made at the PNL decontamination test facility which indicates that the advantages of the low concentration process can be extended to all LWRs.

Compatibility of the decontaminating solution with primary system components must be demonstrated, as well as practical and safe methods of disposing of the spent solutions. A large R&D program involving industrial and academic laboratories will be needed.

- Research programs to develop low concentration decontamination processes for both BWRs and PWRs should be continued and expanded.

2. Operating Transients

Plants are typically shut down about five or more times a year, sometimes for planned reasons, but just as often for unexpected reasons. For the most part the unplanned shutdowns are caused by such events as turbine trip and loss of feedwater. From long experience such events have come to be expected, and little further research is needed to understand them. However, two unusual types of transient do warrant further study: station blackout (i.e., loss of all AC or DC power), and events initiated by complicated interactions between the plant control systems.

a. Loss of Electric Power

The possibility of complete loss of electric power supply, either DC or AC, is a significant contributor to total risk for PWRs. In the larger context of protection of the plant itself, the reliability of electric power is vital to all nuclear power plants, because extended failure would lead to severe damage to the core. Improvement of reliability of electric power, either offsite or onsite, would probably be the most effective way to reduce risk to the public and to the industry. It is therefore somewhat surprising that no research to improve the reliability is included in any of the programs the Review Group has considered.

A reasonable first course would be to review carefully the fault trees that analyze electrical failure, to determine which problems are most important and which are most amenable to remedy. A modest program to improve reliability should evolve naturally, and it could be highly cost-effective in reducing risk. Some possible alternatives that might develop are improved diesel generators, alternatives to improved diesels, portable generators with tie-on locations, instrumentation for DC battery banks, automatic testing for weak cells in battery banks, etc.

- A program should be initiated to develop and evaluate methods of providing more reliable electric station power, both AC and DC. Such a program would have significant value and an excellent cost-benefit ratio.

b. Plant Control System Interactions

The traditional approach to LWR design has been to require that reactor protection systems be "safety grade," but to impose no special requirements on other control systems. In recent years, several severe or potentially severe transients in LWRs have been initiated by control system failures. Also, it has been recognized that, in some cases, control system failure might not only cause a challenge to safety systems but also negate or complicate the efficacy of some of the safety systems needed for the transient under consideration.

Research is needed to better define the role of plant control systems in LWR safety so that those changes which are important to safety can be made.

- Research on the role and importance of plant control systems in LWR safety should be greatly augmented.

3. Other Internal Initiators

a. Fires

Since the fire in the Browns Ferry plant, considerable NRC attention has been aimed at reducing the likelihood that fire will cause serious accidents. There seems to be no need for any expanded research in this area.

b. Sabotage by an Insider

The matter of how and when to include design measures and/or other requirements to protect against sabotage by an insider is complex and requires careful, detailed study if a nearly optimal approach is to be developed. The NRC has completed some scoping studies on this matter. The next step in instituting a systematic and sufficiently detailed examination and evaluation of possible approaches to development and adoption of NRC criteria on this matter should be given the necessary priority in the NRC safety research program.

- The NRC should undertake the studies needed to develop design measures and/or other measures which protect against sabotage by an insider while not compromising safety in other respects.

B. External Initiators

1. Earthquakes

Many probabilistic analyses conclude that earthquakes are an important contributor to the overall risk. The validity of this conclusion is hard to assess because of the large uncertainty associated with earthquake analysis. The uncertainty appears both in estimates of the frequency vs magnitude of large earthquakes and in estimating the probability of damage by earthquakes of a given magnitude. Comments on research needed to address these issues are noted in other sections of this report, particularly in the discussion of piping and probabilistic risk analysis. Estimates of the risk from large earthquakes show a great sensitivity to assumptions as to the size of the largest possible earthquakes in a given seismic region.

- The prediction of earthquake magnitude as a function of frequency still has significant uncertainties and is an important area for further research.

2. Floods

Methods for estimating the flood level as a function of flood frequency are still uncertain.

- Research aimed at developing more realistic predictions of flood level as a function of flood frequency is warranted.

3. Wind

The frequency of tornadoes is well known from weather records. The wind loadings and the impact of wind-driven missiles have been well studied. Further research is not needed in this area.

4. External Fires and Explosions

The effects of fires and explosion from sources outside the plant will generally have to be dealt with on the basis of engineering judgment. No research is needed in this area.

II. Response to Accident Initiators

As implied in the previous section, the first level of defense against accidents is to design, build, and operate the plant so as to reduce the frequency of possible accident initiators. The second level of defense is provided by plant systems designed to respond to initiating events and terminate their effects before serious damage occurs. The plant contains a number of systems designed to respond automatically to a variety of initiating events. Most important among these are the reactor shutdown system, the emergency core cooling system, and the heat removal system. On a longer time scale (greater than about 10 minutes) the operator is also required to respond.

A. The Reactor Shutdown System

All reactors are required to have a quick-acting highly reliable system to stop the chain reaction. This is accomplished by rapid (a few seconds) insertion of neutron poisons called control rods into the core. In addition, LWRs have more slowly acting systems which inject boron-containing liquids into the primary coolant. The effect of these poisons on the chain reaction is well understood. The principal question remaining is just how reliably these systems function. Comments on the need for developing better techniques for estimating the failure rate of highly reliable systems are made in Section V.

B. The Emergency Core Cooling System

Following a rupture of the primary system boundary, it would be important to add water to the system to make up for the water lost out of the break. For large breaks, large volumes of water at low pressure would be required, while for small breaks, small volumes of water at high pressure would be required.

1. Large Loss-of-Coolant Accidents

The large loss-of-coolant accident has been studied in great detail for well over a decade. Elaborate thermal-hydraulic codes have been developed and checked against experimental measurements. The most pertinent full system measurements came from the LOFT tests. There is now little doubt that the current NRC methods of determining the effectiveness of the systems for coping with large loss-of-coolant accidents are very conservative.

The LOFT program to study loss-of-coolant accidents (LOCAs) in a small PWR was initiated in 1962. The program was redirected in 1967, and in 1969 LOFT was redesigned to include features that model the emergency core cooling system (ECCS) of a large PWR. Two nuclear-powered tests of large-break loss-of-coolant accidents had been run by May 1979, showing that the ECCS gave results even more favorable than expected. The facility has since been used to study transients and small-break LOCAs. Thus, the original purpose of the program has been largely fulfilled and the facility has also been useful in working on questions raised by the TMI accident.

The LOFT program has been a success in that it answered satisfactorily the major question of reactor safety for which it was designed, but it does not directly address the key safety problems that are apparent today. The Review Group is reluctant to suggest termination of a facility with such a capable staff and one that represents a large capital investment. But the expense of its operation leads to the conclusion that it is no longer a cost-effective facility. It is unlikely to uncover any major new safety issue. The Review Group has identified a number of high priority areas in which research results are badly needed, and for which the resources saved through phasing out LOFT should be applied.

- The LOFT program should be phased out in an orderly manner. The Re view Group is divided on whether the experimental program for LOFT should be completed in FY-82 or FY-83.

Semiscale is a small mockup of portions of a PWR system. It is versatile and capable of performing experiments on a short-time schedule. It has limitations in size and ability to provide a representative configuration of PWR components, but it is an important facility for carrying out thermal-hydraulics tests. The opinion of industrial representatives is divided on the usefulness of Semiscale, although there is support for a modification that would mock up the once-through steam generator system of Babcock and Wilcox. The cost of operating this system is small enough that we feel its support for the study of thermal-hydraulic phenomena associated with loss-of-coolant accidents should be continued.

2. Small Loss-of-Coolant Accidents

The Reactor Safety Study indicated that the small LOCA is a greater contributor to risk than the large LOCA. The ability to analyze the small LOCA has not kept up with the ability to analyze the large LOCA, in part because of the long computing machine runs required and the resulting buildup of calculational error. In addition, the integral LOCA experiments in the NRC program were designed to test calculations of large LOCAs, and are not well suited to small LOCAs.

The amount of damage to the reactor core from a small LOCA will be very sensitive to the extent and duration of uncovering of the fuel. Some levels of the boiling boundary below the top of the fuel can be tolerated for some time, but the specifics are still more uncertain than is desired. An NRC program to clear up the uncertainty is definitely needed.

- The NRC program on the small loss-of-coolant accident should be continued, to the point where capability to answer important questions of thermal-hydraulics and fuel performance is assured.

C. Decay Heat Removal Systems

Following shutdown of a nuclear plant, the radioactivity in the core continues to generate substantial amounts of heat. To prevent fuel damage, this heat must be removed. The decay heat removal system has this function. Risk analysis indicates that potential failure of this system is an important contributor to the overall risk.

From operational experience, probabilistic risk assessments, and detailed design reviews of specific plants, it is clear that significant improvements in LWR safety could be achieved through improved, more reliable, shutdown heat removal systems. Several countries, including Germany and Switzerland, have required not only highly reliable shutdown heat removal systems but in addition a dedicated bunkered system which provides backup to a loss of all normal offsite and onsite emergency power, and also protection against fire or sabotage. Conceptual design studies of the various principal design alternatives for new and existing LWRs are needed in sufficient detail for development of design and/or performance criteria for improved shutdown heat removal systems.

- A major effort should be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems, for both the reactor vessel and the containment.

D. Operator Response

The correct operator response to unexpected events depends in large measure on his understanding of the system and on how unambiguously the information presented to him portrays the actual situation in the plant. The accident at TMI revealed problems in both these areas, particularly with regard to understanding unusual transients.

1. Engineering Simulation

The Department of Energy has been instructed by Congress to study the possible usefulness of a national reactor engineering simulator facility. Most technical groups have expressed doubt as to the usefulness of such a facility, a view which this Review Group shares. However, the Review Group does favor a program to develop one or more national system analysis facilities for LWRs. Principal emphasis in the program would be placed on the development of methods to improve present capabilities in system modeling, analysis of severe accidents, and ability to compute in real time or faster. Such simulation facilities would not include a full-scale control room mockup, nor would they be intended for emergency response applications or for operator training. Compromises between physical accuracy and computer running time might be necessary to enable the simulation facility to be used to study large numbers of transients. Such important questions as design alternatives and the effects of operator interaction could be explored parametrically.

Such a system simulation facility(s) should be useful for studies by reactor designers, reactor regulators, and the technical support groups of nuclear utilities. The last group could obtain important generic insights, not now available, from simulators or from actual operational experience. Ultimately, this knowledge base would be reflected in the education and training of reactor operators and supervisors.

- A national LWR system simulation program should be undertaken cooperatively by DOE and NRC. This program would treat both PWRs and BWRs generically. The principal goal should be the development of computational capability to study LWR behavior in real time or faster through a wide range of severe transients including accidents involving extensive core damage.

2. Validated Data

The information presented to the operator should be clear and unambiguous. Considerable work is underway by the industry and the NRC on developing effective means for presenting pertinent information to the operator. These are commented on in Section V. However, experience has shown that an important potential source of error occurs when incorrect information is presented to the operator because of instrument error. Significant industry effort is now addressing this issue. There seems to be no NRC work in this area.

One way in which the reliability of information to the operator can be important to safety is illustrated by the transients involving power supply failure to non-nuclear instrumentation at the Rancho Seco, Oconee, and Crystal River plants. The specific source of difficulty in these cases is being remedied, and current development of parameter display systems may alleviate concerns of this kind.

However, other potential scenarios remain, including those associated with a large earthquake near the plant, for which the reliability of information supplied to the operator might have safety implications. Hence, a reasonably comprehensive examination of this subject should be undertaken.

- A broadened research program should be undertaken on the safety implications of the reliability of information provided to the operator.

E. Rulemaking on Minimum Engineered Safety Features

The NRC is planning a rulemaking hearing on minimum engineered safety features. It is important that a detailed review be carried out to assess the information needs for this hearing, and that any research programs needed to provide additional information on a timely basis be identified and started quickly.

- The NRC should promptly identify those information needs of the rulemaking on minimum engineered safety features of future LWRs that can be supplied in time by safety research, and assign the necessary priority and resources to this task.

III. Fuel Damage and Mitigation of Its Effects

The general topic of damage to nuclear fuel as a result of potential accidents is divided into three parts: 1) the fuel damage process itself, 2) the release of radioactivity from the fuel and its transport to the containment boundary, and 3) the effectiveness of the containment at preventing the release of radioactivity under loads caused by various postulated accidents.

A. The Fuel Damage Process

The topic of fuel damage covers a wide range of possible conditions, from perforation of some of the cladding to total melting. Some minor cladding failure and cracking could result from thermal stresses and other effects of normal operation, including anticipated transients. Serious failures of the cooling system could result in various degrees of overheating that would lead to gross cladding failure and eventually to deterioration and melting of the fuel itself.

The possibility of damage to the fuel is a major safety issue because under normal conditions the UO_2 fuel pellets form an effective trap for most radioactive fission products. Thus any large release of these fission products would be preceded by serious damage to the fuel.

1. Normal Operation and Anticipated Transients

Safety-related research on this item is needed to maintain a low probability of fuel element failure despite plans to increase the burnup routinely achieved with LWR fuel. Low failure frequency limits the amount of radioactive material in the primary system. This facilitates maintenance, repair, and in-service inspection, and therefore helps to keep the integrated occupational man-rem dose at a low value. A lower inventory of radioactivity in the primary system would also reduce any offsite release from certain postulated minor accidents, such as steam generator tube rupture. It would reduce some problems in waste handling.

In addition to an extension of the current research and development on pellet-clad interaction to higher burnups, consideration will have to be given to any possibility of change in the release characteristics of fission gas from the UO_2 fuel pellet during anticipated transients in reactors containing higher burnup fuel. Research in these areas is related to work historically conducted either by industry or under DOE and its predecessors.

- Research on fuel element behavior during normal operation and anticipated transients should be the responsibility of the nuclear industry, or be part of a program of generic studies supported by the Department of Energy (DOE).

2. Super-Prompt-Critical Bursts in Power

This was of considerable interest in the 1960's when reactivity insertion accidents in LWRs appeared to have the potential for bringing a significant amount of oxide fuel beyond the melting point, perhaps to the vaporization point, thereby possibly generating damaging pressure pulses and disruption of the core. However, design changes have made reactivity insertion accidents of this magnitude and speed highly improbable in LWRs, and recent analyses have indicated that for some of the reactivity-insertion accidents still receiving consideration, the power burst would be terminated sooner than was previously calculated.

Hence, it does not appear that additional research in this field is required at this time.

3. Local Melting of Fuel Pins at Power

It has been suggested from time to time that if melting of a significant portion of a subassembly were to occur at high power through flow blockage or some other cause, the event might propagate. Research was deemed necessary to ascertain the reality of such a possibility, and the time scale for detection of the fuel melting and any required mitigative action.

The flow blockage issue is essentially confined to boiling water reactors, whose fuel assemblies have side walls isolating each assembly from its neighbors. The General Electric Company has analyzed such an event, assuming that for some reason one full fuel assembly was blocked, and has concluded that timely detection of the event and shutdown of the reactor would occur, with no significant potential for rapid propagation. The possibility of performing experiments in the Power Burst Facility (PBF) to examine the matter further has been proposed, but, as of now, no such experiments have been scheduled. Additional insight may arise from the experimental program related to degraded and molten cores, which is discussed below.

4. Degraded and Molten Cores

Although recommendations that research was needed on phenomena related to degraded and molten LWR cores were made as many as fifteen years ago, it was the accident at Three Mile Island that provided the impetus for initiation of an NRC research program on this matter. The question is complex, since many different kinds of phenomena are involved, and the situation could in principle be caused by many different initiators, each with its own circumstances.

It will be useful to structure the consideration of degraded LWR cores by classifying conceptual accidents into two categories: (1) accidents that would be terminated before a large part of the core had melted (TMI-2, for example), and (2) accidents that would cause the entire core to melt, threatening the integrity of the reactor vessel and subsequent release of the core to the containment building.

The longer term research programs addressing the first category should be developed in light of two important classes of questions:

(1) Could research significantly influence the course of an accident in which choices available to the operator might affect whether the accident is successfully terminated? If so, how? What research might be done? Is the research feasible technically and financially? How might its usefulness be achieved?

(2) Can research on fuel performance prior to extensive melting lead to improved design of new LWRs or changed designs of existing LWRs? If so, how? How reliable and accurate will estimates of the improvement be?

In the more immediate future, programs may be necessary to ensure to the fullest extent possible a base of information for the degraded core rulemaking hearing.

A similar set of questions must be addressed by that part of the research program that considers phenomena related to molten cores:

(1) Could research on fuel behavior in the reactor vessel during a core melt accident significantly influence the management of an actual accident? If so, how?

(2) Could research on fuel behavior in the reactor vessel during a core melt accident significantly affect the design of current or future LWRs? If so, how?

(3) Are there other objectives of such research than accident management or design changes?

Of course, research programs on degraded cores should also be confined to questions whose answers may have real significance, and which can be well enough defined.

The NRC has announced its intention to hold a rulemaking hearing on "Degraded Core Cooling" to consider how degraded cores should be addressed in the licensing process. To prepare for this hearing, the industry has started the IDCOR program, to carefully review and analyze all available pertinent data. This information and possibly results of new experiments are being used to develop improved analytical tools for degraded core issues. It appears that few, if any, results from currently planned NRC research programs on degraded cores can be expected in time for the hearing. Specific examples of areas with limited or non-existing data relevant to degraded cores are factors affecting the extent of core melt, gross clad damage, change in fuel geometry, and coolability of degraded cores. Some additional experimental and analytical studies could still be initiated and conducted in time to be useful for the hearing. Consideration should be given to establishing priorities, with this objective as the highest priority. Care should be taken not just to duplicate work of the IDCOR program.

- The NRC should promptly identify that information which it needs for the degraded core cooling rulemaking and which safety research can supply in time for the hearing. It should assign the priority and resources to get the job done, deferring, if necessary, other less urgent degraded core safety research.

5. Three Mile Island Recovery

A careful experimental study of the seriously degraded core in TMI-2 would provide very valuable information on many aspects of core degradation for use in future analysis of such accidents. It is important that enough support be available to obtain the unique information that can be found by detailed examination of this damaged core. It is recognized that the severe institutional and financial problems impeding TMI cleanup must be solved to make this possible. The longer the inspection of this core is delayed, the less trustworthy and the less valuable will be the results of the examination.

- Ample federal funds should be made available for developing timely information through examining the degraded core in TMI-2. The current program of DOE to carry out such investigations should be supported.

6. Power Burst Facility (PBF)

The PBF is a facility unique in this country for the study of fuel failure mechanisms. It provides a service essential to determining damage to an order of magnitude or better as a precursor to inserting more sophisticated experiments into such facilities as NRU or ESSOR. Unless such scoping studies are done beforehand, approval to insert such experiments in other reactors could be difficult or impossible to obtain. For this reason and because of valuable data from the scoping experiments themselves, continued support of PBF is desirable.

On the negative side, some of the recently proposed programs for PBF appear ill conceived and could lead to misinterpretations. This is particularly true of experiments aimed at extrapolation to full core behavior. Such extrapolations could lead to results which are either much too conservative or much too optimistic. The program now proposed should be reviewed for its relevance and modified accordingly.

- Power Burst Facility operation should continue; however, the proposed NRC programs relevant to degraded cores, etc., should be reviewed critically and modified as necessary.

B. Release and Transport of Radioactive Material*

Understanding of the nature of the risks from potential reactor accidents depends heavily upon estimates of the form and amount of radioactivity that might be released from the reactor containment building under a variety of postulated accident conditions. This area can be subdivided into three parts:

(a) The amount and form of radioactivity released from the fuel itself under various conditions of degradation, which will be called the fuel release process.

(b) The amount of removal of radioactivity by plateout, washout, or agglomeration that takes place during transport from the fuel to the containment barrier, which will be called the radioactive transport process.

(c) The modes of containment failure and their effect upon the fraction of the radioactivity that is released to the environment, which will be called the containment failure process.

1. Fuel Release Process

The risk from reactor accidents is thought to be dominated by those failures that would produce serious degradation of the fuel from overheating or melting. Current estimates are based mainly on the Reactor Safety Study (WASH-1400) which contained analyses performed between 1972 and 1975. The WASH-1400 report assumed that in the most serious core melt accidents, well over 50% of the most volatile fission products would be released to the primary system. It also assumed that up to 10% of the less volatile fission products would be released from the fuel. These assumed fractions were based upon laboratory experiments on very small amounts of UO_2. It was noted in the WASH-1400 study that these values were probably conservative. However, a review of the TMI accident and other fuel failures indicates that during such incidents high release fractions of such volatile elements as Cs and I are not only possible but likely. It has been suggested by some, though, that the physical and chemical form of the released radioactivity can have a large impact on the transport processes. In this regard, the question of whether the iodine is released as elemental iodine or as the compound CsI may be of importance. A second question of importance is the size of aerosol particles released and the aerosol density.

* This topic is often referred to as "the source term." In fact, each accident scenario has its unique source term.

2. The Radioactive Material Transport Process

Radioactive gases and aerosols released from the fuel would have to pass through part of the primary system and then into various compartments in the containment structure to pose an increased threat. During this process, there would be potential for plateout on cold surfaces, agglomeration of aerosols into heavier particles which may settle out, or washout by contact with water. For the very worst accidents considered in WASH-1400, it was assumed that a relatively small fraction (less than half) of the radioactivity released from the fuel would be removed by these processes. The method used for estimating the fraction of radioactivity removed by these processes in WASH-1400 may be conservative for many of the accident sequences. This is important because if the estimate of releases of all the major isotopes other than the noble gases were reduced by a factor of 10 or more, the calculated number of acute fatalities would go to zero in every case, and there would be a substantial reduction in other health effects as well. However, for this to change the current view of the nature of reactor accidents, it would have to be established not only that the removal fraction is larger but that it is larger in all the important accident sequences. Research on this transport process should receive a high priority, and should include analysis of transport of fission products in real accidents that have occurred. The following areas are particularly in need of investigation:

(a) Plateout in the primary system.
(b) Washout by sprays or passage through suppression pools and ice condensors.
(c) Rates of agglomeration of very high density aerosols.
(d) Deposition on wet containment surfaces.
(e) Removal during passage through containment fractures.

The NRC has recognized the importance of this issue and recently issued a report entitled "Technical Basis for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, June 1981, which points out the importance of these removal processes, but concludes that existing knowledge does not warrant major changes in the WASH-1400 values of the removal fractions. In addition, NUS Corporation under contract to DOE is reviewing these issues and suggesting research to resolve them. A report containing their recommendations is to be published shortly. Finally, in preparation for the announced NRC Degraded Core Cooling Hearings, the industry has organized a major review of these issues in their IDCOR program.

> The NRC, DOE, and industry have ongoing programs to provide improved understanding of radioactive releases possible after severe reactor accidents. These should encompass a range of conditions and accident scenarios sufficient for broad understanding of the processes of transport and release of the important radioactive isotopes. The NRC program should provide as much information as possible for the degraded core cooling hearings. There is also a need for a long-range program in this area.

C. Containment Effectiveness

As would be expected, accidents with major offsite consequences can occur only if the containment fails. Thus, the causes of possible containment failure and the failure mechanisms themselves are very important to reactor safety. The failure modes could be of three types:

(a) Failure to isolate.

(b) Failure due to excessive internal pressure from static or dynamic loads.

(c) Failure due to melt-through by the molten core.

Failure to isolate refers to failures of the systems that are intended to seal the containment under upset conditions. These failures could result either from equipment (hardware) failures or from human failures either in maintenance or operation of the system.

Overpressure failures could result from a gradual buildup of steam pressure due to a failure of containment heat removal following a serious system failure. They might result from a sudden pressure surge caused by hydrogen detonation or burning, or from a sudden large steam release.

In analyses such as those in WASH-1400, all very large releases are estimated to be the result of catastrophic containment failure either before or shortly after core melt. On the basis of current knowledge it seems clear that if the containment were to remain effective for several hours or more following core melt, substantial reductions in the release fractions would take place. Thus the timing of containment failure would also be important.

1. Containment Isolation

All containments are designed to isolate as a result of signals generated by a wide range of abnormal conditions. For overall safety this isolation system must be highly reliable. Very careful attention to this issue is expected during the design and licensing of the plant. No areas of large uncertainty associated with this process require specific research programs. However, the general comments on improvements to probabilistic risk analyses apply to analyses done on this system to estimate its reliability.

2. Overpressure Failure

During the past decade, most containment research has been focused on subcompartment dynamic loads for PWRs and dynamic loads in BWR suppression pools as they might arise from steam relief or a LOCA. Most research on the latter has been done by the affected utilities and the reactor vendor, and has progressed reasonably well.

A markedly different kind of containment research has arisen as an aftermath of TMI-2. Partly as a consequence of evaluation of the capability of various types of containment to withstand hydrogen burning, and as a result of the special reviews being given the risk from Zion/Indian Point, the failure mode and pressure are being calculated for a range of types of loading. Failure point estimates are currently being made, on the assumption that the building has been constructed in conformance with the design drawings, although for existing plants test values rather than minimum code values are sometimes used for the strength of steel or concrete.

The Nuclear Regulatory Commission is developing and implementing requirements with regard to hydrogen control capability which frequently take into account the existence and effects of pressures well beyond the normal design point. The rulemaking on degraded cores may lead to a range of new requirements for existing and future reactors. Risk-based analysis is likely to become increasingly important in decision making, and such analyses will surely include the mode, point, and timing of containment failure. A failure of leak tightness seems much more likely than a catastrophic structural failure, but this remains to be assured. It may become important in the estimation of failure probability to allow for the possibility of flaws in fabrication and construction, deterioration due to aging, and design errors. It may become important to determine that pressure below which there is a very high probability of retaining containment integrity.

It appears that it will be necessary to develop a new area of research involving a marriage of the techniques currently used by structural engineers with those employed by reliability engineers. This research will probably be largely analytical since integral experiments are difficult, expensive, and too limited in their range of applicability to be definitive.

- The reactor containment is the ultimate barrier providing protection of the public in the event of a radioactive release from the fuel. The NRC research programs to better understand containment response to loads created by postulated accidents are important and should be supported. A close coupling between structural analysis techniques and the methods of probabilistic risk analysis is needed to make better estimates of the probability of containment failure as a function of increasing internal pressure.

3. Effects of Hydrogen

The possibility of hydrogen generation and release in containments has been a recognized safety issue in water reactors for several decades. Problems with safety are not limited to cases of major fuel failure in a large LOCA but may be associated with several factors, such as rates and mechanisms of formation, concentrations for flammability and explosion, and methods of correction and control.

Questions related to hydrogen were recognized as constituting an early generic safety issue which was considered at one point to be resolved by actions taken in Regulatory Guides.

Recently a generic issue on hydrogen was reestablished and promoted to the limited list of unresolved safety issues. It was one of four issues reported to Congress as new unresolved safety issues in 1981.

This increased concern with hydrogen has resulted from a variety of factors, such as enhanced interest in the effects of degraded cores and the greater implications of hydrogen explosion in smaller containments such as the ice condenser.

The task action plan for attacking the hydrogen problem will be developed this fiscal year, and research programs will be established. This program should be given priority and pursued diligently. Complementary programs by industry should be considered in establishing research scope. The significance of the problem should be established in terms of containment design and volume.

- The NRC is initiating a research program on the issue of evolution and burning of hydrogen in accidents. This program is important and should be pursued expeditiously. Complementary programs by industry should be taken into account in structuring the NRC's research program. The significance of containment design and volume should also be taken into account.

4. Alternative Containment Designs

Alternative containment designs are under consideration that would make the containment more effective at coping with loads imposed by postulated degraded core and fuel melt accidents. The possibility of requiring such measures in future designs will be an important part of the Degraded Core Cooling Rulemaking Hearing. It is essential that prior to the hearing the NRC carry out research on the effectiveness of such measures as containment venting without filtering, containment venting with filtering, and core debris retention in reducing public risk. The effectiveness of such measures compared to other such possible alternatives as the dedicated heat removal system discussed in Section II will be an important issue in the hearing that will require this information.

- The NRC safety research program on mitigation of degraded core and core melt accidents should be modified as necessary to provide the information needed on alternative containment design concepts under consideration, including improved containment cooling, containment venting, venting and filtering, core debris retention, and hydrogen control.

IV. Effects of Potential Accident Releases to the Environment and Their Mitigation

Significant amounts of radioactivity are predicted to be released from the containment in the most serious postulated accidents. It is important to estimate the magnitude and likelihood of the possible consequences of such a release, both as to public health and the damage to property. This information not only provides the basis for risk estimates but is required for establishing the civil protection procedures that might be employed to mitigate the consequences. This section considers the current state of the ability to analyze the consequences of releases of various sizes and types and the impact of proposed mitigating actions.

A. Distribution of Radioactive Materials

Analyses of reactor accidents suggest that two pathways should be considered for the distribution of radioactivity following an assumed accidental release. By far the most important in terms of early health effects and property damage is the release of radioactive gases or fine aerosol particles that would be distributed under the influence of prevailing weather conditions. The second pathway would result from the release of radioactivity into surface or groundwaters.

1. Meteorological Modeling

Most analyses have used a version of a "Gaussian Plume" weather model to calculate the distribution of airborne releases. Many, but not all of these models do not allow for changes in wind direction. The computer code CRACIT developed by Pickard, Lowe, and Garrick includes a detailed treatment of wind changes. All meteorological models used contain numerous simplifying approximations that affect their accuracy; important among these is the treatment of rain. The overall accuracy is probably good for a flat inland site for distances of 5 to 10 miles from the assumed release. However, on sites with special topographical features such as deep river valleys, the local topography introduces features that compromise the accuracy.

The NRC is planning to fund the development of a new improved version of the code CRAC (Consequences of Reactor Accidents Code). The Review Group supports this effort and suggests that particular attention be paid to improving the meteorological modeling. The research should include more detailed weather models than the "Gaussian Plume" model, and particular attention should be given to including improvements in handling effects of rainfall and local topography.

- The methods for calculating the influence of meteorological conditions on the distribution of radioactive material following a postulated accident should be improved. The NRC effort to do this through a new version of CRAC (Consequences of Reactor Accidents Code) should continue to be supported.

2. Hydrological Modeling

In accident scenarios in which it is postulated that the reactor base mat fails following a serious release of radioactive material into the containment, a large amount of radioactive material might enter the groundwater. This released radioactivity could eventually find its way into water supplies and be the source of protracted low-level doses. Because of the dilution by groundwater and the longer time scales for groundwater movement, it is highly unlikely that these doses would contribute to early health effects. However, most risk analyses suggest that such releases might be from ten to a hundred times more likely than a large airborne release. The result is that when the long-term health effects are weighted by their probability, the risk may be of the same order as that from airborne releases. More needs to be known about the distribution of radioactivity by groundwater both for potential reactor accidents and for radioactive waste disposal. It would be reasonable for the NRC to continue work at a modest level in this area.

B. Civil Protection Strategy

All estimates indicate that civil protection actions could reduce significantly the effects of accidental releases of radioactive materials. Considerable regulatory effort has gone into planning of evacuation. Clearly, if people were moved out of the path of the airborne radioactivity they would receive no acute doses. However, if they were in the process of evacuation and failed to get out of the path of the radioactive cloud they would receive higher doses than if they had stayed inside their homes or offices. Thus, for part of the population a strategy of sheltering until the radioactive cloud has passed followed by relocation of those in a contaminated area would be the preferred protective action. The NRC has sponsored work in this general area and the Review Group supports their continuation of this work.

Research in this general area is regarded as important, but the Review Group has not examined the full range of activities by federal, state, and local governments, and therefore makes no recommendations for specific research that may be needed.

C. Biological Effects of Radiation

Since the effects of x rays on radiologists were first recognized, it has been widely realized that radiation in various forms can have serious biological effects. During the last fifty years, extensive research programs have been carried out to determine the quantitative effects on a variety of biological systems including man. Although some aspects of the biological effects of radiation continue to be controversial in scientific circles, these effects are much better understood than those of most of the potentially harmful substances being introduced into the environment by man. The current knowledge is sufficient for reasonable, realistic, or bounding estimates of biological effects of radiation caused by reactor accidents. The largest uncertainty is in the area of the effects of low levels of radiation.

- The national research programs aimed at a better understanding of the biological effects of radiation should be continued at their present level.

D. Economic Impacts of Large Releases

Estimates of the economic impacts of large accidental releases of radioactivity on property show them to be in the same range as large natural disasters. A major component of these costs is from decontamination of land. Very little research activity exists on procedures for decontaminating large land areas, and a modest research program is warranted.

- A modest research program into techniques for decontaminating large land areas is warranted.

V. Important General Issues

In the review of reactor safety, some technical issues were identified that were so broad and so pervasive that they did not conveniently fall into any one of the preceding categories. These include human error and probabilistic risk analysis (PRA). Because of the special importance of these issues, they are discussed in this separate section.

A. Human Error

Most studies of the likely causes of serious accidents conclude that over 50% of the risk is associated with human failure to perform as intended. This observation includes human errors in design and construction, in maintenance and testing during operation, and, of course, mistakes by operators in response to unusual occurrences. The major goal of research in this area is to reduce the incidence of serious human errors. The current program considers three principal ways of doing this: 1) through better training of personnel, 2) through better system design, 3) through improved procedures. Although some proposed improvements would get broad support, many others may be improvements for some events but could reduce the effectiveness of dealing with other events. Thus it is important to be able to judge the overall impact of any specific change in training, procedures, or design on overall system safety. Because of the unpredictability of human behavior it is often difficult if not impossible to do this by probabilistic methods.

A broad research program is needed to improve understanding of the impact of humans on system reliability, operability, and safety, and of other factors that affect the performance of man-machine systems. While human engineering and the man-machine interface have been subjects of particular interest to NASA, DOD, and other industries and agencies for a number of years, the subjects received little attention from the nuclear industry and the NRC until the TMI accident. Since TMI, both NRC and the industry have launched substantial human factors research programs, but the present knowledge base is inadequate and more research is required.

NRC's program plan for human factors research currently includes studies of human error rate, review of control room design from a human factors perspective, and enhancement of operator selection, training, and performance. The Review Group believes that, in general, the work outlined in this plan is useful and important.

However, the accomplishment of many of the objectives would appear to require more resources than are currently assigned to the work. In many cases, the deliverable results of the research are not well defined. Further, in some areas, particularly the human factors aspects of graphic displays and computer aids to reactor operators, the proposed work appears to be overly duplicative of ongoing or completed industry work. The program associated with LOFT appears to be of marginal relevance for this reason.

It is important that the regulatory standards, criteria, and guidance that the NRC must establish in the human factors area be supported by the best possible knowledge base. It is therefore necessary that NRC identify the information it needs for developing criteria, standards, and regulatory guides and revise its human factors research program as needed to obtain that information. In identifying its information needs and revising its programs, the NRC should be cognizant of past and ongoing research programs of NASA, DOD, the nuclear industry, and other establishments. The revised program should be complementary to and not overly duplicative of industry efforts by owners' groups, INPO, EPRI, and NSAC.

Finally, although the Review Group recognizes that identifying valid ways to measure the effectiveness of human performance will be very difficult, it is a critical element that should be addressed in the research program.

- Probabilistic risk analysis indicates that about half the risk from reactor accidents is attributable to human error. The body of knowledge concerning pertinent human factors is inadequate, and it is important that further research be done in this area to provide an adequate technical basis for regulatory activities. The NRC should establish relative priorities, so that its research program will be structured to obtain the most important information first. The important information is that which can be used to improve control room design, operator aids, and selection and training of operators, which should lead to reduction of rates of human error. Further, the NRC program should coordinated with industry efforts.

B. Probabilistic Risk Analysis (PRA)

Since the accident at Three Mile Island the use of PRA as a tool for helping to understand a variety of safety issues has rapidly increased. This use has ranged from estimating the reliability of specific plant systems such as the auxiliary feedwater and scram systems, to complete integrated estimates of overall plant risks similar to that done in WASH-1400, the Reactor Safety Study. The Review Group supports this use of PRA techniques as one of the tools for reaching a better understanding of plant risks and the reliability of specific systems.

It must be recognized, however, that the techniques are relatively new and therefore there is room for considerable improvement. Three areas needing improvement are: 1) input data on basic failure rates; 2) treatment of common cause failures; 3) analysis of highly reliable systems.

In many cases the failure probabilities required as input data for any quantitative analysis are subject to considerable uncertainty. This can be the result of limited input data or of inapplicability of the general data to the system under analysis. The data on human failure rates comprise just one of a number of cases where considerable uncertainty exists. Significant reseach is needed to develop a more reliable data base.

The term common cause failures refers to two or more failures that occur simultaneously as a result of some common cause. This can be the result of an external event such as a fire, earthquake, etc., or of a common design defect. The normal analysis of a system first looks for single random failures or combinations of single random failures that can lead to system failure. The result of this is called the probability of failure by random events. The analyst must then go back and look for possible dependencies between these previously assumed random failures, and the result is the probability of failure due to common causes. In well-designed systems where the probability of failure by random component failure is calculated to be quite low (less that one in 10,000 per demand), it is customarily found that common cause failures are dominant contributors to system failure. Furthermore, the various sources of these common cause failures are often subtle and hard to identify. Numerous techniques are used to estimate the impact of common causes on system failures, but it is commonly recognized that much more research needs to be done in this difficult area.

The analyses of very well-designed systems sometimes predict failure rates in the range of one in one million (10^{-6}) to one in one billion (10^{-9}) per demand. Although it is possible that such high reliability can be achieved, experience has shown that more often than not such predictions tend to be overly optimistic. Very unlikely events tend to be more frequent than these low probability values suggest. For example, a system out of service for maintenance one minute each year has an unavailability of 2×10^{-6} just for that cause. Natural events that may be as unlikely as one in a million years could overwhelm probability estimates that otherwise are very low. Another area where probability estimates tend to be highly uncertain is the probability of damage to systems subjected to large ground motions. Much more work is needed in the area of estimating the failure probabilities of very reliable systems.

At a time when the number of analyses being requested by the NRC is increasing rapidly, it becomes very important to standardize as much as is practical the techniques to be used. If done with care, this should improve the quality and reproducibility of the results.

The NRC currently has research programs addressing each of these issues and because of the importance of PRA techniques these programs should be expanded.

- The use of probabilistic analysis is rapidly expanding. A significantly larger research program is needed to improve and standardize the methodology as much as possible. Particular areas that need improvement are: methods of handling common cause failures, data collection and analysis, accuracy of the estimated frequency of very low probability events, and assurance of the quality of analyses.

VI. Organizational and Institutional Comments

A. NRC Safety Research

The NRC's safety research program evolved to its present form over an interval of about eight years. This form is now responsive to needs as they developed over that period. The program is not logically or optimally structured in coverage and emphasis to respond to questions of currently perceived urgency. The importance of the loss-of-coolant accident in the historical development of the NRC's regulatory methods has led to focus of research on this topic, with most of this work devoted to the large LOCA. It seems appropriate at this point to change the emphasis of the program. Some of this change is underway, and suggestions are made throughout this report for additional changes.

The program heavily emphasizes reduction or prevention of consequences once an accident is assumed to have started, since the regulatory staff has itself adopted this primary objective. As said below, this emphasis seems appropriate as a guideline, but it should not be rigid. On numerous occasions, NRC research has been necessary to improve reactor safety, and without doubt more such instances will arise.

Most of the NRC's programs are strong, and together they constitute one of the more important research programs in the country, in size, importance, and product. The program is broadly accepted by the technical community, especially where it is more fully understood and its impact is greatest. The program has substantially improved the understanding of many of the reactor safety issues it has explored, and it has greatly improved the underpinning of NRC's regulatory actions on light water reactors.

The Review Group found a widely held perception among reactor safety researchers that the results of research do not lead to corresponding changes in regulations, particularly in those cases in which the results showed the regulations to be based on conservative assumptions. Even after discounting the tendency of researchers to feel their results are not used enough, this criticism appears to be essentially valid. Three examples related to Appendix K to 10CFR50, which resulted from the ECCS Rulemaking hearings in 1972-73, stand out. Appendix K has not been altered in spite of a considerable body of experimental data from the NRC's own research programs showing several of the prescribed features to be very conservative. Chief among those is perhaps the decay heat formulation commonly referred to as ANS + 20%. This formula, to which the analysis is sensitive, is high by a good 25%. Similarly, data obtained since Appendix K was issued show that the zirconium oxidation rate in steam is less than predicted by the prescribed formulation. Finally, the integral experiments conducted on LOFT indicate that the clad temperature rise in a large-break LOCA will be hundreds of degrees below the value computed by the Appendix K methodology.

Although it is probably true that the NRC staff frequently incorporates new data into their engineering judgments, more formal incorporation of new and significant results is desirable. It is true that the NRC frequently faces industry (vendor or utility) opposition, active or passive, in making regulatory changes even when the proposed change would relax the rules. The industry tends to grow "comfortable," or it learns to live with a particular regulation. The prospect of the procedures involved in a change gives more concern than is compensated by the prospect of a relaxed specification. There is a tendency to retain the outdated assumptions in regulatory actions to provide extra overall safety margin, even when the conservatism has become assured.

The NRC program is in many areas weak in timely production of results, partly because of inflexibility and partly because the requirements for new reseach were not recognized sooner. The inflexibility has several origins, which are addressed in a recommendation below.

A basic criticism heard many times was that the Commission fails to set and communicate policy direction to the staff. There is even a lack of general instructions to the research staff as to what areas of research are considered important. In the absence of direction, the staff does not have an adequate basis to set research priorities and establish schedules which conform to the responsibilities faced by the Commission. This has been one of the causes of the uncertainty in goals that has been observed. The Review Group did not make any in-depth study of this organization problem. However, various individual members found such criticism to conform to their own observations. It goes without saying that an adequate solution to this problem is vital to an adequate reactor safety program, both inside and outside the NRC.

The planning and management of research have probably been diluted by the recent reorganization that combined the Office of Nuclear Regulatory Research and the Office of Standards, On the other hand, the Office of Nuclear Regulatory Research has never become well integrated with the licensing staff, and the new reorganization may ease this problem.

- The Commission should encourage and promote the visible integration of the results of safety research into the regulatory process. Regulatory requirements should not rest on bad science when good science has become available. If retention of conservatism is desired because of uncertainties or for other reasons, this should be done through the application of explicit safety factors to a calculation based on best available methods, in accordance with good engineering practice.

- The limitation of NRC's programs to "confirmatory research" should be removed, so that exploratory research and research to improve safety can be undertaken when this looks like the better course to follow.

- It is reasonable that the prevention of accidents should be the principal area for industrial safety research, and prevention or limitation of public consequences the principal area for NRC. This difference should be retained as a guideline but not as an absolute boundary between the programs.

- The flexibility of the NRC's research program should be improved. An easier way to transfer funds from one decision unit to another should be established. The methods of inter-office coordination and concurrence on new programs should be simplified to prevent a single individual or Office from being able to block programs that have broad support. Subject only to Commission agreement, the Office of Research should be permitted to undertake research that is expected to be important in the long run, even if there is no immediate user need. Of course, this should not displace research for which a user need has been defined and agreed on.

- NRC's long-range research plan (NUREG-0740) should be restructured to follow lines of Agency objectives. This should infuse it with a more up-to-date logical structure and discourage the tendency to support doing more of what has already been done. Introduction and continuation of guidance at the Commission level will be of fundamental importance in confirming the objectives and restructuring the plan. The restructured long-range plan should be more specific in defining the deliverables in each program.

B. The Role of DOE in Light Water Reactor Safety Research

Public Law 96-567 (Nuclear Safety Research, Development, and Demonstration Act of 1980) states that: "a proper role of the Federal Government in assuring nuclear power plant safety, in addition to its regulatory function, is the conduct of a research, development, and demonstration program to provide important scientific and technical information which can contribute to sound design and safe operation of these plants." It is the intent of the law that this program be carried out by the Department of Energy.

The DOE maintains a number of national laboratories which are a reservoir of talent and facilities for conducting reactor safety research and design studies. There is little doubt that a valuable program can be formulated to make effective use of these resources. PL 96-567 specifies a number of areas in which the DOE should establish research programs. Although the Review Group does not believe that DOE is necessarily the best agency to conduct each of the types of research specified, there will probably be enough flexibility to permit DOE to formulate a sound program. However, DOE does not now have the staff structure necessary for planning and conducting this program. In response to the passage of PL 96-567, the Office of Nuclear Power Systems of DOE has conducted a study of the kind of program it would propose and has prepared a draft of the report to Congress. The Review Group found the first draft of this report to contain too many generalities to give much confidence that a well-focused research program will be formulated.

One section of PL 96-567 calls for a study of the need for a "reactor engineering simulator facility." The DOE appears to be reaching a negative conclusion regarding such a need, a finding in which the Review Group concurs. The Review Group does believe, however, that there is need for more detailed and more physically accurate mathematical models of nuclear power plants and rapid calculational methods that incorporate these models, a need discussed elsewhere in this report.

One research project which should especially receive DOE support is the study of the damaged core in the Three Mile Island, Unit 2 plant. DOE has already initiated such a project. This core is a national resource in the sense that it contains more information about the progression of fuel damage during an accident and the characteristics of degraded cores than could be obtained practically in a costly planned research program. The DOE program to study the TMI-2 core is supported in a recommendation elsewhere in this report.

Other areas of safety research which the Review Group finds somewhat more appropriate to DOE than NRC include improved steam generator design, increased reliability of nuclear plant AC and DC electrical systems, and development of more reliable valves.

- DOE should form a strong staff of technical and managerial personnel knowledgeable and experienced in the subject of water reactor safety, to develop and implement programs of safety research in this field.
- DOE should develop a program of generic research to improve water reactor safety, and assume a substantial responsibility in the area of accident prevention, to supplement other programs designed to reduce the likelihood of nuclear power plant accidents.

C. Research in Other Countries

Research on safety of nuclear power plants is extensive in some other countries, especially in Japan and West Germany. These programs provide an effective supplement to American research. The participation of the NRC and industry in foreign programs (e.g., Halden, Marviken, LOCA programs) has been beneficial, as has foreign participation in U.S. programs (e.g., LOFT, PBF). Interaction with research programs in other countries has been effective in preventing unnecessary duplication of work. DOE on its entry into water reactor safety research should also begin to take advantage of research done abroad.

- DOE should assume a more active role in the international aspects of water reactor safety research.

D. Freedom of Publication

Publication and communication of results of water reactor safety research programs by NRC, EPRI, and foreign countries have generally been good. A steady stream of reports is issued in this field, compared to a bare trickle about ten years ago. However, proprietary considerations severely limit the dissemination of results of research by reactor vendors and utility owners' groups.

In recent years, the publication and distribution of results of DOE's research on reactor-related subjects have been restricted. This policy, which was adopted in order that results of the research could be used to trade for results of related research by other countries, is mostly applied in the fast breeder field. Research on water reactor safety should be treated differently in this respect from developmental reactor research. Freedom of flow of information should be the rule, in recognition of the fact that reactor safety everywhere benefits all.

- DOE's water reactor safety research program should follow a policy of free and open publication and distribution of reports. However, this policy should permit protection of proprietary information of industry participants.

E. The Role of Risk Assessment

Probabilistic risk assessment can in principle be used to rank programs on a cost-benefit basis. It is important that this practice be started. However, other factors must also be used in arriving at such a ranking. Among these are timing of the program compared with timing of research needs, the probability of success, the impact the research may have, and the extent of resources required.

- A policy should be instituted whereby probabilistic risk assessment is used as one tool in establishing priorities of research programs.